# THE NUCLEAR REACTOR CLOSED CYCLE.

**John K. Sutherland.**
*Fredericton, New Brunswick, Canada.*

**Keywords**

Nuclear reactor cycles, closed cycle, enrichment, depleted uranium, fuel fabrication, maintenance wastes, fission nuclides, spent fuel, reprocessing, advanced reactor cycles, breeder reactor, retired weapons.

**Contents**



**Glossary**

| | |
|---|---|
| AECL | Atomic Energy of Canada Limited |
| AGR | Advanced Gas-cooled Reactor |
| BNFL | British Nuclear Fuels Limited |
| BWR | Boiling Water Reactor |
| CANDU | CANadian Deuterium (natural) Uranium |
| CANDU-6 | The CANDU reactor of about 680 MW gross output |
| CSFR | Commercial Scale Fast Reactor |
| DPFR | Demonstration or Prototype Fast Reactor |
| DU | Depleted Uranium |
| DUPIC | 'Direct Use of Spent PWR fuel in CANDU' |
| EBR | Experimental Breeder Reactor |
| EFR | Experimental Fast Reactor |
| FBR | Fast Breeder Reactor |
| GCR | Gas Cooled Reactor |
| GW(e) | Gigawatt (electrical) |
| HEU | High Enriched Uranium |
| HLW | High Level Waste |
| HWLWR | Heavy Water Light Water Reactor |
| IAEA | International Atomic Energy Agency |
| ISL | In-Situ Leach |
| kWh | kilowatt hour |

| | |
|---|---|
| LILW | Low, Intermediate Level Waste |
| LWR | Light Water Reactor |
| MeV | Million Electron Volts |
| MOX | Mixed Oxide Fuel |
| MWh | Megawatt hour |
| n,∀ | Neutron captured, alpha particle emitted |
| n,p | Neutron captured, proton emitted |
| n,( | Neutron captured, gamma energy emitted |
| n,2n | Neutron captured, two neutrons emitted |
| NEA | see OECD NEA |
| NNPT | Nuclear Non-Proliferation Treaty |
| NEI | Nuclear Energy Institute |
| OECD NEA | Organisation for Economic Co-operation and Development - Nuclear Energy Agency |
| PHWR | Pressurized Heavy Water Reactor |
| PWR | Pressurized Water Reactor |
| RBMK | Russian graphite moderated reactor (Chernobyl type) |
| SWU | Separative Work Unit |
| TU | Transuranium - unstable elements above uranium in atomic number |
| TWh | Terawatt-hour |
| VVER | Russian Pressurized Water Reactor |

**Summary**

This article examines the stages of the closed nuclear reactor cycle from mining of the ore through to spent fuel in preparation for reprocessing and disposal of fission wastes. There is a discussion of fission, activation and transuranium nuclides and their half-lives and abundances. The various options of dealing with spent fuel - reprocessing or not - are examined from both a technological viewpoint as well as from the political actions in the U.S. which have so far denied the U.S. nuclear industry the Fast Breeder Reactor and the reprocessing option. Some aspects and outcomes of this political decision are examined. The impact of reprocessing spent fuel or not, on radioactive waste volumes, and the subsequent management and security timeframe, is examined. The current retirement and destruction of military nuclear warheads from the U.S. and former U.S.S.R. arsenals is briefly touched upon in the use of MOX fuel in existing nuclear reactors. The avoidance of proliferation risks from increasing quantities of non-reprocessed spent fuel in the future because of contained transuranium nuclides, and the assurance of relatively non-polluting energy sources for the future, require that the Fast Breeder Reactor program and reprocessing, be re-implemented in the near future. In addition, the breeder cycle opens up the immense energy potential contained in surface stockpiles of depleted uranium, and allows it to be used rather than wasted. Breeding also expands the commercially valuable natural uranium resource by allowing lower grade deposits to be exploited including uranium in seawater, and also opens up the vast resource of energy contained in the much more abundant thorium-232.

## 1. The Closed Nuclear Cycle

### 1.1 Introduction.

The nuclear reactor cycle, from uranium mining to final waste disposal, comprises several stages. These stages depend upon the reactor design and type, and whether or not spent fuel is re-processed - (the 'closed nuclear fuel cycle'); stored ('once-through'); or the reactor operates with some combination or modification of these cycles. The closed nuclear fuel cycle is becoming increasingly used in some European countries and Japan.

The use of breeder reactors based upon uranium or thorium, requires a closed nuclear fuel cycle in order to return the bred transuranium nuclides, uranium-233 and unused uranium and thorium back into the cycle.

Some of the stages in the fuel cycle are associated with the production of various classes and volumes of radioactive wastes throughout the world as shown in Table 1.

**Table 1. Summary of Accumulated Radioactive Wastes and Non-wastes in the World to about 2000.**

| Reactor Cycle Stage | Radioactive Wastes | Non-wastes for recycling |
|---|---|---|
| **Front End** | | |
| Uranium Mining | 1 000 000 000 + tonnes | |
| Processing | Minor | |
| Refining | Minor | |
| Conversion | About 35 000 $m^3$ | |
| Enrichment | About 16 000 $m^3$ | |
| Depleted Uranium | 'Waste' – only if not recycled | 1 500 000 Tonnes |
| Fuel Fabrication | About 160 000 $m^3$ | |
| **Back End** | | |
| Spent Fuel | 'Waste' – only if not reprocessed | 230 000 Tonnes |
| Maintenance LILW Wastes | About 6 000 000 $m^3$ | |
| Spent Fuel Reprocessing | | 218 000 + Tonnes |
| Fission Wastes (4 percent) | About 9000 + Tonnes | |
| **Military** | | |
| Retired Weapons HEU U-235 | | 500 Tonnes ± (U.S. & Russia) |
| Retired Weapons plutonium-239 | | 500 Tonnes ± (U.S. & Russia) |

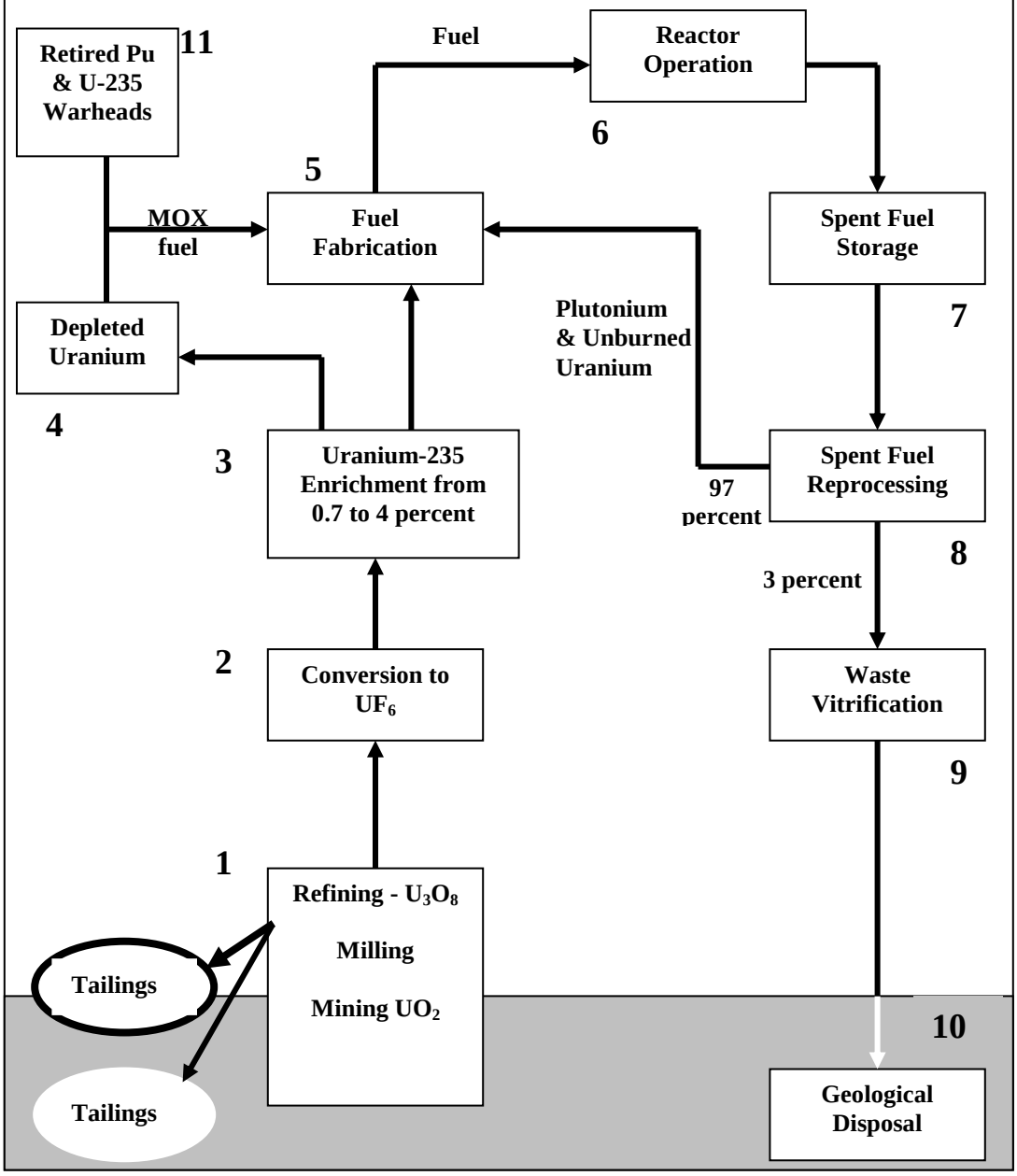
The closed-cycle stages from mining to final disposal are shown in Figure 1. The first 8 stages are described in this article. The remaining stages are described in more detail in Article 3.6.3.6 dealing with nuclear waste disposal and the destruction of military 'wastes' - including retired weapons - in the reactor cycle.

The processes leading up to loading fuel into the reactor are known as the 'front end' of the cycle, and those following discharge of spent fuel from the reactor are known as the 'back end' of the cycle.

The simpler, but more resource-intensive, 'once-through' cycle foregoes - for the moment - the re-processing option and the associated relatively low volumes of wastes. Without reprocessing, the entire discharged spent fuel, containing about 97 percent unused uranium and transuranium elements, may be required to be managed as long-term waste.

With reprocessing, only the low volume, relatively short half-life, fission nuclides would be managed as waste. The 'once-through' cycle requires that world uranium mining production be maintained at a relatively high level to keep up with the demand for new fuel. Where natural uranium is used in the reactor (e.g., the CANDU), uranium enrichment is not required and reprocessing is not considered at this time, as replacement fuel is cheap at about U.S.$22 to $30 $kg^{-1}$, relative to the costs of enriched fuel, and relative to the costs of reprocessing.

**Figure 1. The Closed Nuclear Fuel Cycle for a Typical PWR or BWR Reactor**



## 2. Uranium Mining, Processing, Refining

The first mining efforts to deliberately recover uranium occurred in central Europe to extract uranium for use in coloring glass and glazes. When radium - one of the radioactive progeny of uranium - was discovered in 1897 and immediately became of value in medical radiation treatments, a mining boom of known uranium-bearing deposits took place, with the uranium itself being treated mostly as a by-product or waste.

Mining methods of economically viable deposits may be by open pit (about 38 percent at the end of 2000), underground mining (about 33 percent), in situ leaching (ISL)(about 17 percent), or as a by-product of other mining or industrial process (about 12 percent).

By-product uranium is recovered from activities such as phosphate mining and processing for fertilizer production; formerly, from the processing of some alum shale deposits in Sweden; formerly, from some low-grade coal deposits in the U.S.; and from some gold and copper mines. Increasingly, more uranium deposits at the present time are exploited by in situ leaching of the deep ore body to extract uranium, which is then pumped in solution to the surface for extraction. This method produces neither rock waste nor tailings. Where the ore is mined, rather than chemically leached, it is crushed at the mine site, reduced to sand-sized particles, leached with a solvent solution, and then is further processed to extract and purify the uranium.

The residual wastes from mining the common low-grade deposits (ranging mostly from about 0.1 percent to 1 percent uranium) amount to large quantities of rock and process tailings containing traces of residual uranium too difficult to extract, and radium (of no significant value today) along with most of its progeny. Such wastes today amount to more than about 200 million tonnes in surface waste piles in the U.S. alone, and possibly ten times more at existing and former uranium mining operations throughout the world. Most are now gradually being addressed to ensure that they are adequately covered and protected to delay, and thus minimize radon gas leakage from them; to limit moisture penetration and acidic drainage; and to protect them from weather erosion. Extraction of uranium by in situ leaching avoids most of these problems. Modern mining is much more stringently regulated and controlled than previously, with ongoing environmental protection and remediation requirements and activities.

The richest uranium deposits at the present time are found in Canada and Australia which dominate world production (see Table 2). The richest ores are the primary oxides of uranium (uraninite and pitchblende) - usually in mineralized veins with other metals, such as silver, copper, bismuth, cobalt, molybdenum and lead as sulfides, selenides, tellurides and arsenides along with various oxides and silicates. Many of the richest known deposits have generally been mined out, with a few notable exceptions in Canada and Australia. There are also many secondary and very complex uranium-vanadium minerals - often brightly colored green and yellow - which tend to be more widely dispersed through the sedimentary strata in which they are found, as in Colorado, and in many other low grade uranium deposits throughout the world.

| Table 2. Estimated Recoverable World Uranium Resource at US$80 kg$^{-1}$ of Uranium | | |
|---|---|---|
| | **Tonnes** | **Percent** |
| Australia | 890 000 | 26 |
| Kazakhstan | 560 000 | 17 |
| Canada | 510 000 | 15 |
| South Africa | 350 000 | 10 |
| Namibia | 260 000 | 8 |
| Brazil | 230 000 | 7 |
| Russia | 150 000 | 4 |
| United States | 125 000 | 4 |
| Uzbekistan | 120 000 | 4 |
| Niger | 70 000 | 2 |
| Ukraine | 45 000 | <1 |
| Others (28 countries) | >50 000 | 1 |
| | | |
| Total* | 3 360 000 | |

* At 41 000 tonnes a$^{-1}$ production, this estimated resource will last for less than 100 years at this price, without reprocessing, and without the adoption of the fast breeder cycle.
Data are from various sources.

The extraction of uranium from seawater (even at about 3 parts per billion, or 3 milligrams per tonne of seawater) is a possible future massive source of uranium. In Japan, the Takasaki Radiation Chemistry Research Establishment conducted extraction experiments several years ago and determined that uranium and other metals could be readily extracted from seawater, but not yet economically at the prevailing price for uranium. The Uranium contained in the Black Current flowing off the coast of Japan carries about 5 million tonnes of uranium (comparable to the estimated remaining uranium resource on land) each year along the coastline. Japan uses about 6000 tonnes of uranium annually, of which only about 3 percent is actually consumed in the reactors during each fuel cycle, so extraction of even a small fraction of that carried in this current, could meet their needs as well as those of the rest of the world. With reprocessing and the use of a fast breeder cycle the resource is essentially unlimited.

## 2.1 Ore processing, Concentration and Refining

This converts the extracted and purified uranium to $U_3O_8$, also known as yellowcake. This is a pure, but very low-level radioactive material. It is traded internationally and safely shipped around the world in 100 L steel drums to uranium enrichment facilities, or may be fabricated into natural uranium fuel for use in those reactors (CANDU and GCR) fueled by natural uranium.

The world production of uranium in the year 2000, controlled by 8 major mining companies operating in about 16 countries, was about 41 000 tonnes of $U_3O_8$. With an average grade of about 1 percent $UO_2$ in the feed ore, this implies that more than 4 million tonnes of radioactive mine wastes are produced annually from these deposits. Comparable radioactive wastes are produced from many other base metal mining operations, to the extent of about 1 billion tonnes each year. Most of these other mine wastes are controlled primarily to minimize acid mine drainage effects and erosion, and attract little attention because of their contained radioactivity, which is usually unmeasured, uncontrolled, and ignored.

The largest producers of uranium for sale on the international market are Canada and Australia (annually about 11 000 and 8000 tonnes respectively in 2000) producing more than 50 percent of the world supply. Kazakhstan appears to be making a major effort - announced in 2002 - to eventually become the dominant producer in the world.

## 3. Conversion to $UF_6$

Conversion is the process of changing $U_3O_8$ (yellow-cake), to uranium hexafluoride $UF_6$ - a reactive white crystalline solid - for enrichment in the uranium-235 isotope.

There are five commercial conversion plants in the world: in the U.S., Canada, France, the United Kingdom, and Russia. Two other countries, Brazil and China also operate relatively small conversion facilities but not, at present, commercially. Total available capacity in the seven facilities to the end of 2000, is about 66 000 tonnes/a, but annual world requirements for conversion are approximately 57 000 tonnes.

There are only minor low-level uranium wastes associated with such conversion. The cumulative total of such wastes throughout the world up to the year 2000, amounts to about 35 000 $m^3$.

## 4. Enrichment

Natural uranium contains 99.3 percent U-238 and 0.7 percent U-235. Nuclear fission reactors based upon uranium, cannot operate without the uranium-235 isotope, and in the case of light water moderated reactors, require the concentration to be greater than about 3 percent.

Enrichment is the process of augmenting the percentage of uranium-235 in uranium hexa-fluoride ($UF_6$) and rejecting a stream of uranium-238 (also known as depleted uranium), before the U-235-enriched uranium is processed into the oxide fuel for use in the reactor. Some uraniferous wastes are produced during this process, with world cumulative totals up to the year 2000 amounting to about 16 000 $m^3$.

The two uranium isotopes cannot be separated chemically but have slightly different masses (about 1.3 percent difference), so are physically separable though with considerable difficulty.

There are two common multi-stage enrichment processes - gaseous diffusion, used for the Manhattan Project in the 1940s, and gaseous ultra-centrifuging in Calutrons - with others (laser ionization coupled with magnetic separation) being researched. The process, taking into account the market price of uranium and the high electrical energy cost of enrichment (described in Separative Work Units – SWUs – the amount of electrical energy needed to produce 1 kilogram of enriched uranium), still leaves about 0.25 - 0.3 percent U-235 in the rejected uranium-238.

The total U.S. nuclear electrical capacity of about 100 GW(e) from more than 100 large reactors requires some 12 million SWUs per year to enrich the fuel. Each SWU - using the gas diffusion process - requires about 2500 kWh of electricity or the equivalent of about 4.2 percent of the total nuclear electrical output (late 1990s). Competing offshore interests, seeking to break into the lucrative U.S. enrichment market, suggest that they can achieve this separation at a much lower cost. Future advances in isotope separation in the U.S., and lower separation costs, may make it economically advantageous to re-

process some of the stockpiled depleted uranium to strip out more of the residual U-235 if the adoption of a breeder reactor cycle and spent fuel reprocessing continues to be politically rejected or remains economically unattractive at the present price of uranium.

To produce about 4 percent uranium-235 enrichment from 0.7 percent feed material requires an almost 8 fold concentration. For every tonne of Low Enriched U-235 produced for the Light Water Reactor (4 percent U-235), about 7 tonnes of depleted uranium (about 99.7 percent U-238) is rejected from the process. For every tonne of High Enriched Uranium (say 20 percent U-235 - the minimum enrichment used in nuclear submarine and ship reactors) about 39 tonnes of depleted uranium is rejected.

In general, the more enriched the uranium, the greater the reactivity margin to over-ride the effects of fission poison build-up, the smaller the required fuel load to maintain a large power output, and the more compact the reactor, as in nuclear submarines and ships.

Uranium enrichment - an expensive and technologically demanding process - was initially a virtual monopoly of the U.S. The early reactor programs of most other countries were based upon the U.S. PWR or BWR reactor designs and U.S. enriched fuel. Other countries either accepted this as the price to be paid for nuclear co-operation and development, or began to develop their own independent enrichment programs, or sought to build reactors that were fueled by natural uranium (as in the U.K. and Canada).

Commercial 'enrichment' (as opposed to 'conversion') is carried out in the U.S., France, the United Kingdom, Germany, the Netherlands and Russia. These countries effectively control the enriched uranium fuel supply to many other countries which operate Light Water reactors. All of these countries must be signatories of the Nuclear Non-Proliferation Treaty (NNPT) and allow International inspection of nuclear facilities, materials and operations to ensure that there is no clandestine diversion of restricted materials. Other countries with enrichment facilities for their own programs include China, Japan, and Pakistan. There are still a few states like India, Pakistan, Israel and North Korea, that possess nuclear facilities and nuclear weapons, but resist signing the NNPT or threaten to withdraw from it (N. Korea withdrew in 2003).

**5. Depleted Uranium**

Depleted Uranium (DU) is defined as pure uranium containing less than 0.7 percent of uranium-235 (and therefore more than 99.3 percent uranium-238) and is the byproduct of the enrichment process. The world production of depleted uranium is currently about 47 000 tonnes a year.

The world stockpile from the last 50 years of enrichment to the end of 2001, and shown in Table 3, amounts to about 1.5 million tonnes of depleted uranium, with about 600 000 tonnes in the U.S. This U.S. DU stockpile has a present-day value of about 100 trillion dollars if it were used for electricity production. This cannot be utilized, however, with the present U.S. political inertia with regard to advanced and breeder reactors.

| Table 3. Estimated World Inventory and Value (if used in the breeder cycle) of Stockpiled Depleted Uranium (2001) | | | |
|---|---|---|---|
| | | | |
| **Country Or Enrichment Company** | **2001 Inventory, (Tonnes)** | **Estimated Annual Increase (Tonnes)** | **Chemical Storage Composition** |
| | | | |
| US | 590 000 | 20 000 | $UF_6$ |
| France | 207 000 | 12 000 | $U_3O_8$ |
| Urenco (UK, Germany, Netherlands) | 53 000 | 4000 | $UF_6$ |
| UK (BNFL) | 30 000 | 0 | $UF_6$ |
| Russia | 490 000 | 10 000 | $UF_6$ |
| Japan | 5600 | 500 | $UF_6$ |
| South Africa | 2200 | 0 | $UF_6$ |
| China | 26 000 | 1000 | ? |
| Others | < 1000 | ? | ? |
| | | | |
| **Total** | **1 404 800** | **47 500** | |
| **US$ present energy value as electricity** | **US$250 trillion** | **US$8 trillion** | |
| Most of the basic data have been revised from original DOE data. | | | |

The depleted uranium is surface-stored in steel canisters, mostly as $UF_6$, to maintain the possibility of revisiting these stockpiles as there is the possibility of utilizing advanced enrichment techniques to access the 0.25 to 0.3 percent of uranium-235 that remains in the depleted uranium. The amount of uranium-235 that is left in the DU is a function of the cost of natural uranium and the cost of a Separative Work Unit. If the uranium feed cost is low or the cost of an SWU is high, then it is cheaper to reject the DU when it still contains about 0.3 percent uranium-235. If the uranium feed cost is high (or the cost of an SWU is low), then more uranium-235 is extracted, before the DU is rejected from the process. Without the development of the breeder reactor cycle, most of this DU cannot be brought back into the reactor cycle and may possibly be managed as nuclear waste. This may happen even though the energy potential in these useable stockpiles is many times that so far obtained from the once-through fission reactor cycle of the entire world in total since the beginning of the nuclear power program.

The calm tolerance of such massive waste in any other endeavor, and if widely publicized, would be cause for great political discomfort, and an immediate re-evaluation, but not yet in the nuclear world.

A few non-energy uses of DU are: as radiation and biological shielding around medical sources and as radiation shielding for HLW shipments. Uranium is denser than lead, and though slightly radioactive, it is a better radiation shield. Other uses include making high density concrete (radiation shielding); counterweights in aircraft; ballast in yachts; and as tips to armor piercing projectiles.

However, its most rational and economically attractive use is as future reactor fuel. It can be used to 'down-blend' HEU from retired weapons-grade HEU, or used as mixed oxide (MOX) reactor fuel when admixed about 16:1 to 25:1 with plutonium-239, available either from retired military plutonium warheads (U.S. and Russia), or from reprocessed spent fuel. In this way, the entire potential energy in the uranium-238 that might

otherwise be discarded as waste becomes accessible by breeding the uranium-238 to plutonium-239, which is an even better fissile nuclide than uranium-235.
The "Megatons to Megawatts" agreement between the U.S. and Russia, initiated in the early 90s, was designed to reduce nuclear weapons stockpiles from the levels reached in the Cold War era. The intent was to convert highly enriched - bomb-grade - uranium from surplus Russian and U.S. warheads into low-enriched fuel for use in commercial nuclear power reactors. In 2002, about half of the more than 100 commercial reactors in the U.S. were fueled with uranium from about 6000 former Russian nuclear weapons warheads. It may seem strange to consider that the potential energy contained in these former weapons, rather than being released in conflict in milliseconds over selected targets is now being released in peacetime in a much slower and controlled reaction, and to the direct commercial benefit of both the U.S. and Russia. The converted warhead materials used in this way, become essentially unrecoverable for anything other than continued energy production.

The process of using former weapons materials, whether uranium or plutonium in a commercial reactor cycle is the only effective and worthwhile way to render these materials unusable for weapons construction, and reduce proliferation possibilities while producing immense quantities of needed electrical energy and avoiding the pollution effects of utilizing fossil fuels for the same energy.

## 6. Fuel Fabrication

Fuel fabricating facilities are located in most countries that operate a significant number of nuclear power plants (see Tables 4, 5 and 6).

| **Table 4. Fuel Fabrication Facilities in the World (2000)** | | |
|---|---|---|
| **Countries Fabricating Light Water Reactor Fuel (Tonnes/a)** | **Countries Fabricating Heavy Water Reactor Fuel (Tonnes/a)** | **Countries with Mixed Oxide Fuel Fabrication Facilities (Tonnes/a)** |
| United States (3900) | Canada (2700) | France (140) |
| Kazakhstan (2000) | South Korea (400) | United Kingdom (128) |
| Japan (1674) | India (270) | India (50) |
| Russia (1620) | Argentina (160) | Belgium (37) |
| France (950) | Pakistan (20) | Japan (10) |
| Germany (650) | | |
| Sweden (600) | | |
| Belgium (500) | | |
| South Korea (400) | | |
| United Kingdom (330) | | |
| Spain (300) | | |
| Brazil (100) | | |
| China (100) | | |
| India (25) | | |
| Pakistan (?) | | |
| **Total 12 299 Tonnes** | **Total 3560 Tonnes** | **Total 365 Tonnes** |
| The data were derived from various internet sources | | |

The world total cumulative low level wastes associated with fuel fabrication, up to the year 2000, are estimated to be about 160 000 $m^3$; a remarkably small volume considering the associated energy production. This is approximately equivalent to the volume of a 100 by 50 by 30 metre arena.

| Table 5. Nuclear Power Plants in Commercial Operation (early 2002) | | | | | | |
|---|---|---|---|---|---|---|
| **Reactor type** | **Countries** | **Number** | **GW(e)** | **Fuel** | **Coolant** | **Moderator** |
| | | | | | | |
| Pressurized Water Reactor (PWR, VVER) | US, France, Japan, Russia, & most others | 259 | 231 | Enriched $UO_2$, MOX | Water | Water |
| Boiling Water Reactor (BWR) | US, Japan, Sweden, Germany | 91 | 79 | Enriched $UO_2$ | Water | Water |
| Gas-cooled Reactor (GCR & AGR) | UK | 34 | 12 | Natural U, enriched $UO_2$ | $CO_2$ | Graphite |
| Pressurized Heavy Water Reactor "CANDU" (PHWR) | Canada, South Korea, Argentina, India, Romania, China | 34 | 16 | Natural $UO_2$, PWR spent fuel, MOX | Heavy water | Heavy water |
| Light Water Graphite Reactor (RBMK) | Russia, Lithuania | 17 | 13 | Slightly enriched $UO_2$ | Water | Graphite |
| Fast Breeder Reactor (FBR) | Japan, France, Russia | 3 | 1 | $PuO_2$, $UO_2$, DU (MOX) | Liquid metals | None |
| Other (HWLWR) | Japan | 1 | 0.1 | Slightly enriched $UO_2$ | Water | Heavy water |
| | **TOTAL** | **439** | **352** | | | |
| Source: Nuclear Engineering International and others. Thorium-based breeder reactors have been researched since the 1940s, including a U.S. uranium-thorium HTGR (helium-cooled) experimental reactor at Fort St Vrain - now retired, and are still being researched in several countries, notably in India. The total number of reactors in operation by late 2002 was 442, with 35 more under construction. (IAEA). | | | | | | |

| Table 6. Ten Largest Consumers of Nuclear Power | | |
|---|---|---|
| **Country** | **No. Units** | **Total MW(e)** |
| | | |
| **USA** | **109** | **99 784** |
| **France** | **56** | **58 493** |
| **Japan** | **59** | **38 875** |
| **Germany** | **21** | **22 657** |
| **Russian Federation** | **29** | **19 843** |
| **Canada** | **22** | **15 755** |
| **Ukraine** | **15** | **12 679** |
| **United Kingdom** | **12** | **11 720** |
| **Sweden** | **12** | **10 002** |
| **Republic of Korea** | **10** | **8170** |
| | | |
| **Total** | **335** | **297 978** |
| **World** | **439** | **354 416** |
| Most Data are from the IAEA (2001). | | |

Reactor fuel is made from small cylindrical pellets of pure natural or enriched uranium oxide sintered at more than 1400°C. The pellets are encased in small diameter metal tubes - usually zirconium-niobium alloy - which are arranged into a fuel assembly, which may be a single small cylindrical natural uranium fuel bundle (about 60 cm by 10 cm) as used in a CANDU reactor (using more than 4000 bundles of about 20 kg each) or a relatively large rectangular enriched-uranium fuel assembly in a PWR (about 200 fuel assemblies, each weighing about 700 kg) or BWR reactor. The dimension of the fuel pellets and of the overall fuel assembly are dictated by the

basic reactor design to ensure defined operating and heat removal characteristics and fuel stability over a wide range of operating conditions.

## 7. Reactor Operation, Maintenance Wastes, and Spent Fuel

There are about 443 large operating civilian reactors in the world today (early 2003), producing about 17 percent of the world's electricity. There are about another 35 under construction - many in India - and a further 30 or so that are in various stages of planning. Each operating reactor, depending upon its design, has an annual fuel requirement and annual spent fuel (HLW) discharge rate of between about 20 tonnes (PWR) and 150 tonnes (CANDU). Associated operational and maintenance wastes (Low and Intermediate Level Wastes) make up between 100 $m^3$ to about 1000 $m^3$ each year at each reactor. There are no significant atmospheric emissions of any kind and all solid wastes are controlled and managed.

### 7.1 Maintenance Wastes.

Reactor maintenance activities produce various radioactive wastes. These include solid and non-compactable materials which may be internally contaminated with fission nuclides. These may include discarded and replaced valves, piping, pump parts and some ventilation components. They also include solid or semi fluid spent purification resins which cannot be regenerated, and which are usually relatively highly radioactive, and metallic sludges and oxides from reactor-side boiler cleaning.

Reactor maintenance activities may also generate compactable and non-compactable cleaning wastes such as radioactively-contaminated protective clothing and coveralls that cannot be laundered, cleaning rags, discarded respirators, air filters, plastic sheet, canvas floor covering, plastic and metal containers, contaminated tools, wood, mop heads, and temporary wiring and lighting. Most of these wastes are lightly contaminated and require management for only a few months or at most a few years before they can be redirected to normal waste disposal areas. Some can be recovered and reused.

The Low and Intermediate Level Wastes (LILW) associated with the operation and maintenance of a 1000 MW(e) reactor consist of between 100 and 1000 $m^3$ of managed waste each year. They are usually stored and managed on the reactor site in secure and shielded areas. The bulk of the wastes is made up of very low-level radioactive materials.

The actual volume depends upon the nature and duration of the reactor maintenance work, the effectiveness of waste screening and sorting, and whether or not the wastes are compactable. Typically, they contain minor quantities of relatively short-lived fission products - such as zirconium-95 - from opened reactor systems and relatively short-lived activation products.

These LILW wastes are usually managed at the reactor site in an accessible, secure location (typically shielded in concrete cells) and can be specifically revisited after 15 or 20 years of radioactive decay. They may then be re-classified for either continued

storage; for sorting and re-packaging; or may be incinerated or discarded into normal waste processes as exempt wastes.

**7.2 Spent Fuel.**

Spent fuel is the most radioactive material that is produced during reactor operation and may range from about 20 tonnes up to about 150 tonnes produced in a year of operation of a large commercial reactor. All work activities with fuel, whether fresh or spent, and refueling operations, are stringently controlled and ensure that no individual can directly encounter unshielded spent fuel. Fortunately, it is a relatively low volume material and is easily shielded by several metres of cooling water, which also allows spent fuel inspection and movement underwater into storage trays within hours of being discharged from the reactor. Refueling operations usually take place only about once every 12 to 18 months or longer for PWR and BWR reactors. Exceptions are encountered in those reactors which are continuously re-fueled at power, such as the CANDU, in which one or more of the hundreds of fuel channels are sequentially and remotely de-fueled and re-fueled most days of the week throughout the year, and when the reactor is at power.

The world total of spent reactor fuel (HLW) to the end of 2001 is about 230 000 tonnes. This is added to at the present time at the rate of about 15 000 tonnes per year from all operating reactors.

Fission products from operation of the nuclear reactor are physically trapped in the matrix of the fuel and increase with burn-up of the fuel in the reactor. As some are strong neutron absorbers, they eventually increase to a point where they significantly compete for the limited number of neutrons available for fissioning; begin to over-ride the available margin of reactivity; and begin to internally close down ('poison') the reactor. When this might occur depends directly upon the percentage enrichment and reactivity margin of the starting nuclear fuel in the reactor environment. More highly enriched fuels have a longer burn-up life due to their much greater margin of reactivity to over-ride the growing effect of such neutron poisons.

Fuel placement within the reactor core also strongly affects the rate of burn-up. Centrally placed fuel assemblies are exposed to a higher neutron flux than those at the reactor periphery and reach their target burn-up sooner. At some point - determined by reactor fueling specialists and economics - this reactor-poisoning effect requires that some of the high burn-up fuel be discharged and replaced with new fuel, usually accompanied by fuel re-arrangement within the core, where some low burn-up fuel assemblies from the periphery may be moved closer into the reactor core.

Regardless of the reactor design or degree of enrichment, the fission and transuranium nuclides in the fuel are initially a function of the uranium burn-up, usually expressed in terms of MWdays $tonne^{-1}$. 'Initially', because most fission nuclides have an extremely short half-life and rapidly decrease in activity once the in-core fission process ceases.

For PWRs, the target fuel burn-up has progressed from less than 20 000 MWdays tonne$^{-1}$ in the early years of the nuclear program, to about 33 000 to 45 000 MWdays tonne$^{-1}$ (producing 3 to 4 percent fission waste in the spent fuel) but is now approaching the 40 000 to 60 000 MWdays tonne$^{-1}$ range (4 to 6 percent fission waste). For natural fuel in the CANDU reactor, 'burn-up' is about 7800 MWdays tonne$^{-1}$ (about 1 percent fission wastes), though with some modifications, including using slightly enriched fuel or recycling PWR fuel (DUPIC), this can be increased to about 20 000 MWdays tonne$^{-1}$. By the time spent fuel is discharged from any reactor, about 40 percent or more of the energy in the reactor has been derived from fissioning of plutonium-239, with additional energy contribution from fissioning of other transuranium nuclides.

## 7.3 Fission, Activation and Trans-Uranium Nuclides.

There are about 700 fission, activation, and transuranium nuclides, all of which are undergoing radioactive decay from the moment that they are formed. The fission nuclides, categorized by half-life, are summarized in Table 7. A detailed listing of all fission nuclides, and a graph showing the bimodal mass distribution of fission nuclides from uranium-235 fission, are contained in Appendices 1 and 2.

**Table 7. Summary of Fission Product Nuclides**

| Fission-product Half-lives | Number of Defined* Fission Nuclides |
|---|---|
| | |
| Less than 24 hours | 438+ |
| 1 day to 1 year | 42 |
| >1 year to 10 years | 4 |
| > 10 years | 12 |
| Stable fission isotopes | 101 |
| Total fission nuclides | 615+ |
| * Many fission nuclides have extremely short, and difficult-to-define half-lives. | |

### *7.3.1 Fission Nuclides.*

There are about 615 fission nuclides (most are listed in the appendix 1) with many others that are difficult to define because of extremely short half-lives.

Of these 615, about 450 have half-lives ranging from seconds, up to about 24 hours, and rapidly decay from the spent fuel once the fission process is curtailed. They are no longer present after about 1 month at most, following shutdown, or removal from an operating reactor. A general rule of thumb is that after 10 half-lives, any radionuclide is essentially completely decayed even though it may still be detectable beyond this time.

Because of the rapid decay of these short-lived nuclides, the heat output of the fuel falls correspondingly. Within about 1 second of the cessation of induced fission, either due to a reactor trip or by removal of fuel from an operating reactor fuel channel in the case of the CANDU reactor, the core heat output falls from about 100 percent to about 7.5 percent - the heat output from decaying fission nuclides. After 24 hours, fission decay heat is down to about 0.71 percent of full power heat output, and is continually falling. Within about 10 days, all of the short-half-life radionuclides have decayed to stable nuclides, and thus have disappeared from the spent fuel, and heat output from decay of the remaining longer half-life nuclides is about 0.3 percent of that at full power operation.

About 42 fission radionuclides have half-lives which extend from about 1 day, up to about 1 year. All of these decay totally from spent fuel between about 10 days up to about 10 years following discharge of the spent fuel.

Four fission radionuclides have half-lives between 1 and 10 years. These are: Ru-106 - 1.02 years; Sb-125 - 2.76 years; Pm-147 - 2.62 years and Eu-155 - 4.73 years. These four persist from about 10 years up to about 50 years, at most, before they have essentially decayed away.

**Table 8. Fission Radionuclides with Half-lives greater than 10 years (in order of half-life)**

| Fission Radionuclides * | (Fission yield percent) | Half-life (years) |
|---|---|---|
| Krypton-85 | 1.319 | 10.7 |
| **Strontium-90** | **5.8** | **29** |
| **Cesium-137** | **6.19** | **30.07** |
| Tin-121 | 0.013 | 55 |
| Samarium-151 | 0.419 | 90 |
| Tin-126 | 0.059 | 1E5 |
| Technetium-99 | 6.1 | 2.13E5 |
| Selenium-79 | 0.045 | 6.5E5 |
| Zirconium-93 | 6.35 | 1.5E6 |
| Cesium-135 | 6.54 | 3E6 |
| Palladium-107 | 0.146 | 6.5E6 |
| Iodine-129 | 0.54 | 1.57E7 |

The fission yield percentage refers to the total of all of the fission nuclides with this mass number, and not to the individual radionuclide.
* Radionuclides beyond Cs-137in this table, have either low fission yield, have low energy emissions, or are so long-lived as to be low radioactivity.
Data are from the Chart of the Nuclides.

Of the initial inventory of about 615 fission radionuclides, only 12 (about 2 percent of them) are relatively long-lived fission nuclides with half-lives greater than 10 years. These 12 are shown in Table 8. Of these, only strontium-90 and cesium-137 (less than 0.5 percent of the initial fission nuclides in spent fuel) are significant radiological hazards.

These two radionuclides soon became the only significant radionuclides of concern around the Chernobyl accident site, as the other fission nuclides of shorter half-life had mostly been removed from the environment by decay processes. The others which are longer lived than these two are of less radiological concern because of low energy beta and gamma emissions; because of low yield; because they are generally not encountered in typical pathways leading to humans or other biota; or because they have sufficiently long half-lives to be relatively radiologically harmless.

About 100 fission nuclides are stable.

After the fission nuclides have decayed from spent fuel, the only remaining radionuclides are the relatively low radioactivity and very long half-life uranium, and the surviving transuranium nuclides. The radioactivity of the spent fuel after about 300 or so years of decay, and containing only uranium and fairly long half-life transuranium nuclides, is little different from that of the natural uranium from which it was fabricated, as can be seen from Table 9 data and Figure 3.

| Table 9. Approximate Activities of The 'Longer-lived' Significant Fission and Trans-Uranium (TU) Radionuclides in PWR Spent Fuel (burn-up of about 30 000 MW days tonne$^{-1}$) * | | | | |
|---|---|---|---|---|
| **Nuclides** | **Half-Life** | **Activity/Tonne U after 150 days * of cooling (Bq)** | **Activity/Tonne U after 100 years of storage (Bq)** | **Activity/Tonne U after 500 years of storage (Bq)** |
| **Fission Nuclides** | | | | |
| Niobium-95 | 35 days | 2E16 | 0 | 0 |
| Strontium-89 | 50.5 d | 4E15 | 0 | 0 |
| Zirconium-95 | 64 d | 1E16 | 0 | 0 |
| Cerium-144 | 284.6 d | 3E16 | 0 | 0 |
| Ruthenium-106 | 1.02 year | 2E16 | 0 | 0 |
| Cesium-134 | 2.06 y | 8E15 | 40 | 0 |
| Promethium-147 | 2.62 y | 4E15 | 1E4 | 0 |
| **Strontium-90** | **28.78 y** | **3E15** | **2.7E14** | **1.8E10** |
| **Cesium-137** | **30.07 y** | **4E15** | **4E14** | **4E10** |
| **TU nuclides** | | | | |
| Curium-242 | 162.8 d | 6E14 | 0 | 0 |
| Plutonium-241 | 14.4 y | 4E15 | 3E13 | 1.4E5 |
| Curium-244 | 18.1 y | 9E13 | 2E12 | 4.4E5 |
| Plutonium-238 | 87.7 y | 1E13 | 4.5E12 | 1.9E11 |
| Americium-241 | 432.7 y | 7E12 | 6E12 | 3E12 |
| Plutonium-240 | 6.56E3 y | 2E13 | 2E13 | 1.9E13 |
| Americium-243 | 7.37E3 y | 6E13 | 6E13 | 5.7E13 |
| Plutonium-239 | 2.41E4 y | 1E13 | 1E13 | 9.9E12 |
| Plutonium-242 | 3.75E5 y | 5E10 | 5E10 | 4.99E10 |
| * After reprocessing, which can take place after about 150 days of cooling after discharge, only the fission nuclides, representing about 3 percent of spent fuel, need to be discarded as wastes. The data have been adapted from various sources. | | | | |

### *7.3.2 Activation Radionuclides.*

Activation nuclides are radioactive atoms that can be produced by various neutron reactions: (n, (), (n,p), and (n,∀) with reactor components. A fairly comprehensive list, with their half-lives, is presented in Table 10.

These are formed mostly in the materials within and around the reactor core exposed directly to neutrons. In the case of heavy water coolant, photo-neutrons are produced from deuterium nuclei which absorb very high energy photons (above 2.21 MeV) from nuclear fission, capture gamma emitted after radiative capture, from fission products, and from very short half-life activation products like nitrogen-16 which can be circulated in the coolant heavy water of a CANDU reactor, and can thus appear outside of the reactor core.

They are usually of relatively short half-life and are typically addressed from the point of view of radiation protection procedures during maintenance work and under operating conditions, than at any other time in the life of the reactor. The longer-lived activation nuclides such as tritium, cobalt-60 and carbon-14 are the only ones that also need to be considered during decommissioning activities, as all of the shorter-lived nuclides decay away entirely before reactor decommissioning would be considered. Both tritium and carbon-14 can be fairly easily recovered or disposed of, as neither of them presents a

significant external radiation problem - being pure low energy beta emitters - but cobalt-60 (half-life 5.27 years) also has high energy gamma emissions, and constituents containing significant cobalt-60 impurity must be shielded and safeguarded for up to about 50 years before being recycled.

**Table 10 Main Activation Radioisotopes Produced by Neutron Activation of Reactor Components and other Materials Exposed to Neutrons.**

| Radio-isotope | Source and Reaction | Half-life |
|---|---|---|
| Nitrogen-17 | Oxygen-17 (n,p) | 4.17 seconds |
| Nitrogen-16 | Oxygen-16 (n,p) | 7.13 seconds |
| Oxygen-19 | Oxygen-18 (n,() | 26.9 seconds |
| Aluminum-28 | Aluminum-27 (n,() | 2.25 minutes |
| Argon-41 | Argon-40 (n,() | 1.83 hours |
| Manganese-56 | Iron-56 (n,p) | 2.58 hours |
| Copper-64 | Copper-63 (n,() | 12.70 hours |
| Sodium-24 | Sodium-23 (n,() | 14.95 hours |
| Tungsten-187 | Tungsten-186 (n,() | 23.9 hours |
| Phosphorus-32 | Phosphorus-31 (n,() | 14.28 days |
| Rubidium-86 | Rubidium-85 (n,() | 18.65 days |
| Chromium-51 | Chromium-50 (n,() | 27.7 days |
| Iron-59 | Cobalt-59 (n,p) | 44.51 days |
| Cobalt-58 | Nickel-58 (n,p) | 70.88 days |
| Tantalum-182 | Tantalum-181 (n,() | 114.43 days |
| Zinc-65 | Zinc-64 (n,() | 243.8 days |
| Sodium-22 | Sodium-23 (n,2n) | 2.60 years |
| Iron-55 | Iron-54 (n,() | 2.73 years |
| Cobalt-60 | Cobalt-59 (n,() | 5.27 years |
| Tritium (H-3) | Deuterium (n,() | 12.33 years |
| Carbon-14 | Nitrogen-14 (n,p) | 5715 years |

Most data updated from 'Radioactive Wastes', former U.S. Atomic Energy Commission publication in the series 'Understanding the Atom' and Chart of the Nuclides.

Although tritium is an undesirable activation nuclide that is increasingly formed in heavy water reactors, and to a much lesser degree in all reactors, it has immense value in fusion research and future fusion reactors and is commercially recovered from CANDU reactors for this research use as well as for commercial use, especially in self-powered EXIT signs.

No attempt is made to make commercial use of the relatively low quantities of cobalt-60 in activated reactor components. However,cobalt-60 is commercially produced in certain CANDU reactors where rods of cobalt-59 are introduced into the core flux for about 1 year, before being removed for processing to recover the produced cobalt-60. It is used in radiation therapy devices to kill cancers, saving tens of thousands of lives each year and in irradiation facilities which sterilize hospital supplies. It is becoming increasingly used in food irradiation facilities to eliminate the pathogens from fish, meat and poultry, responsible for most food poisoning episodes which cost billions of dollars of economic loss and thousands of lives each year. Cobalt-60 is also infrequently encountered in scrap yards, and occasionally in metal consignments to and from steel mills, mostly in third world countries. This source of cobalt-60 is generally from improperly retired medical devices which contain extremely high concentrations of the nuclide. There have been a few severe radiation injuries and fatalities associated with these exposures to the raw cobalt-60 in scrap yards and the homes of workers, as in Brazil and Thailand, but if the sources are not detected before they are re-worked and diluted into steel, or soon afterwards, the health implications to the broader population are essentially minor and not detectable, though are highly publicized. Such steel shipments and products are re-called, if discovered.

Neutron activation analysis uses neutron activation - usually in materials introduced briefly into the reactor core, or by exposure to a manufactured neutron source - as an extremely sensitive analytical tool for the detection of trace impurities that might otherwise be very difficult to detect. It is especially valuable in forensic science and in detecting art forgeries.

Many materials exposed deliberately or incidentally to neutrons can become activated. These include the metals used in the reactor construction and other materials in the core of the reactor. Certain absorber rods, used to control the reactor as well as used to produce commercial cobalt-60 become activated in this way. Others include the activation of some fission products within the fuel as well as some gaseous components of air (argon and nitrogen) within reactor spaces as well as some dissolved constituents of cooling water. Certain neutron-absorbing chemicals which are dispersed through the reactor coolant or moderator, (where separate - as in the CANDU reactor) and which are used in reactor control (e.g. gadolinium), may be activated in this way.

Some of the more undesirable and longer-lived activation products such as cobalt-60 and carbon-14, are usually easily avoided by carefully selecting reactor component materials for their purity, or by excluding specific trace impurities that may create a radiological problem for workers during maintenance activities or during decommissioning.

Common activation nuclides in air include argon-41 (half-life 1.83 hours, from the neutron activation of argon-40, forming about 1 percent of all air) and carbon-14 (half-life 5715 years, mostly produced from nitrogen-14). Production of argon-41 and carbon-14 can usually be avoided by using carbon dioxide or helium as a gas in those systems or reactor penetrations that encounter neutrons. Those activation nuclides in reactor metals may include cobalt-60 (half-life 5.3 years, from cobalt-59). Those in water, especially heavy water, include tritium (half life 12.32 years from activation of deuterium) and activation of those salts used for water conditioning (sodium and phosphates) or others (e.g., gadolinium nitrate) which provide neutron shim in the control of the reactor or are used for reactor fast shutdown. With the obvious exception of carbon-14, they are usually of relatively short half-life.

### *7.3.3 Transuranium Nuclides.*

Transuranium (TU) nuclides are those above uranium in atomic number. They are sequentially produced by neutron capture, beta decay and other transitions, in uranium fuel in the operating reactor as outlined in Figure 2.

The production rate and fate of any of these transuranium nuclides is dependant upon four competing processes: spontaneous fission, thermal and fast neutron induced fission, neutron activation, and radioactive (beta) decay.

A listing of some of them showing their thermal fission cross sections; activation (neutron capture) cross sections and their spontaneous fission half-lives are shown in Table 11.

Many of them are fissionable or fissile and eventually contribute significantly to the power output from the reactor.

**Table 11. Significant U and TU Nuclides with indications of their Spontaneous Fission (SF) half-life, (Followed by Thermal Neutron Fission (f), or Activation (Neutron Capture) (() Cross Section.**

| Nuclide | Spontaneous Fission (SF) Half-Life (years) | Fission (f) and Activation (() Cross Sections (Barns) | Half-Life in Years |
|---|---|---|---|
| **Uranium-234** | 1.5E16 | f 0.006 ( 99.8 | **2.46E5** |
| **Uranium-235** | 1.00E19 | **f 584** ( 98.8 | **7.04E8** |
| **Uranium-238** | 8.2E15 | f 1E-5 ( 2.7 | **4.47E9** |
| Neptunium-236 | ? | f **2770** ( 701 | 1.55E5 |
| Plutonium-238 | 4.75E10 | f 17.9 ( 540 | 87.7 |
| Plutonium-239 | 8E15 | f **747** ( 270.3 | 2.41E4 |
| Plutonium-240 | 1.14E11 | f 0.06 ( 289 | 6.56E3 |
| Plutonium-241 | 6E16 | f **1012** ( 361.5 | 14.35 |
| Plutonium-242 | 6.77E10 | f 0.002 ( 19 | 3.75E5 |
| Plutonium-244 | 6.6E10 | f ? ( 1.8 | 8.0E7 |
| Americium-241 | 1.2E14 | f 3 ( 600 | 432.7 |
| Americium-242m | 3E12 | f **6409** ( 1254 | 141a and 14ms |
| Americium-243 | 2E14 | f 0.12 ( 78 | 7.37E3 |
| Curium-243 | 5.5E11 | f **617** ( 130.2 | 29.1 |
| Curium-244 | 1.32E7 | f 1 ( 15 | 18.1 |
| Curium-245 | 1.4E12 | f **2001** ( 346.4 | 8.5E3 |
| Curium-248 | 4.15E6 | f 0.37 ( 2.57 | 3.48E5 |
| Curium-250 | 1.13E4 | f 0.002 ( 0.4 | 9.7E3 |
| Californium-249 | 8E10 | f **1666** ( 504.5 | 351 |
| Californium-250 | 1.7E4 | f 4.09 ( 1779 | 13.08 |
| Californium-251 | ? | f **4935** ( 2878 | 898 |

A relatively large (f) or (() cross section indicates that the nuclide soon fissions or is neutron-activated in the reactor respectively. Spontaneous fission occurs continuously. Data are from the Chart of the Nuclides and other publications.

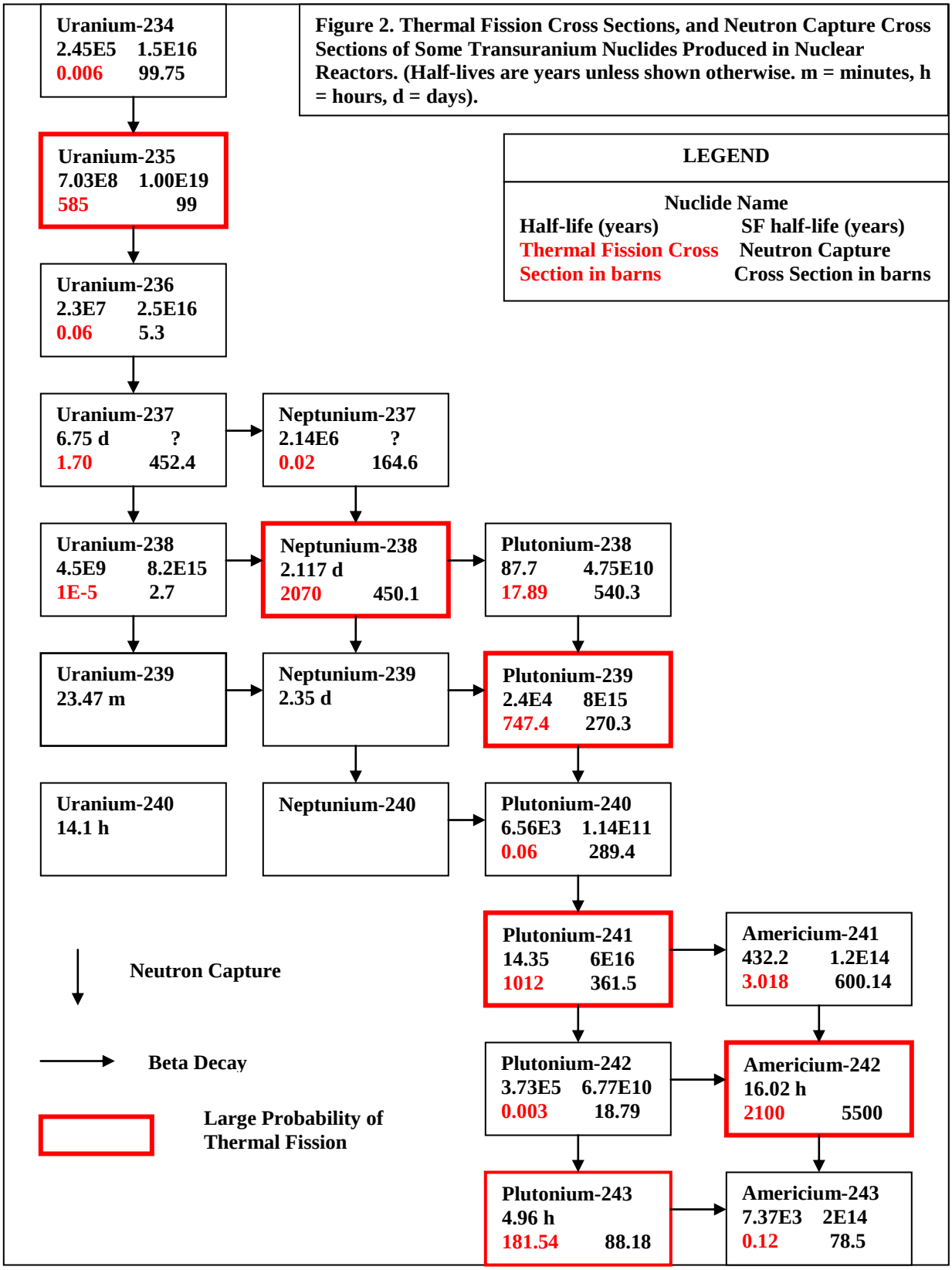



**Figure 2. Thermal Fission Cross Sections, and Neutron Capture Cross Sections of Some Transuranium Nuclides Produced in Nuclear Reactors. (Half-lives are years unless shown otherwise. m = minutes, h = hours, d = days).**

Most TU nuclides are of relatively long half-life, and thus are of relatively low radioactivity compared with fission nuclides, at least for the first hundred or so years as shown in Figure 3. Their presence raises the issue of proliferation, and it is this factor of security that dictates a much longer management time-frame for spent fuel if it is not reprocessed, as well as more stringent requirements for long-term disposal. Such disposal must also allow for the option of legitimate retrieval by future generations to recapture the massive quantities of contained, and unused energy.

In spent fuel, after about 500 years, when the radioactivity remaining in fission nuclides would be almost negligible, the longer-lived TU nuclides remain. Though of low radioactivity, their possible diversion into nuclear proliferation programs is the main reason for maintaining long term spent fuel security and management.

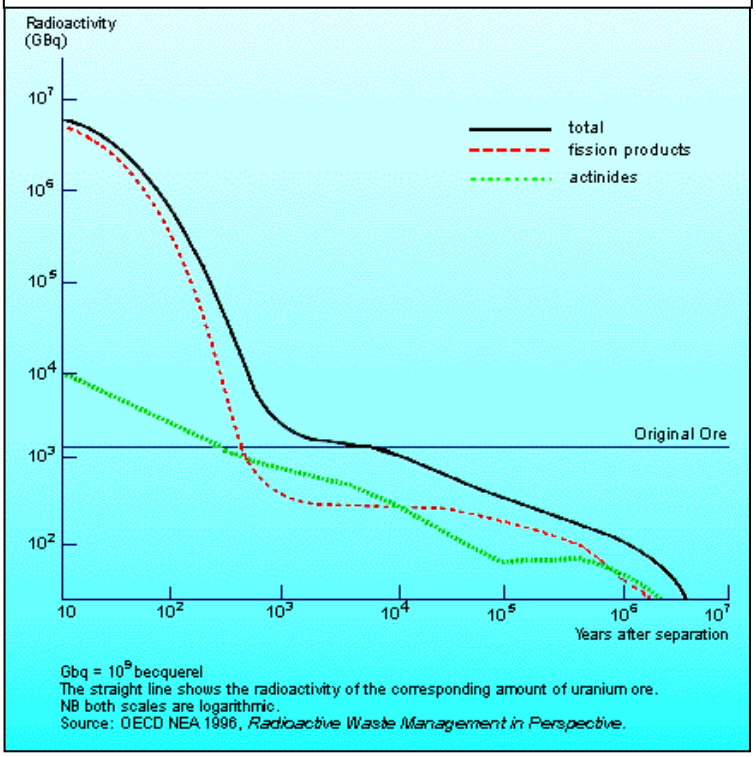
With fuel re-processing, most of the 96 percent residual uranium and the 1 percent of longer half-life actinides including plutonium are removed from the spent fuel and recycled back into the nuclear fuel cycle where most of them contribute to the fission energy. Many of them have a significant fission cross section or a radiative capture cross-section that causes them to transmute into progressively more massive, and occasionally fissionable, nuclides, as shown in Table 11, and Figure 2. The larger the cross section, the more probable is that particular interaction. For example californium-250, with low thermal fission cross section but with a neutron capture cross section of 1779 barns, is preferentially transmuted to Cf-251, which has a notably large fission cross-section of 4935 barns. Some few atoms of Cf-251 escape fissioning by being successively transmuted to heavier californium nuclides which can also be fissioned. Many of them also fission spontaneously and contribute to the reactivity of the core.

Such spontaneously fissioning impurities in spent fuel is one of the major reasons why the attempt to use high burn-up spent fuel as a source material of plutonium for nuclear weapons is extremely undesirable relative to using pure plutonium-239. The inherently unstable (spontaneous fission) impurities and their constantly changing concentrations make the desired reaction unpredictable, difficult to control and much less effective. With reprocessing, volatile fission radionuclides such as krypton-85 and iodine-129 are discharged to atmosphere or, in the case of radio-iodines, may be chemically trapped and managed.

The resulting highly radioactive fission-waste volume to be managed for the longer term - no more than about 500 years - is only about 3 to 5 percent of process-throughput, and the waste conditioning and final management process is very much simplified. This initially liquid waste from the reprocessing cycle is dried and may be mixed with special concrete or with various silicates, boro-silicates and fluxes, before being fused into a solid glass or ceramic block for permanent non-retrievable geological disposal.

Without re-processing, the entire spent fuel charge consisting of more than 95 percent of unused fuel is required to be managed as High Level Waste that requires long term consideration beyond about 500 years, even though the fission nuclides have significantly decayed away by then, and are even less radioactive than the original uranium ore, as shown in Figure 3.

**Figure 3. Decay in Radioactivity of Separated Fission and Actinides from Reprocessing 1 Tonne of PWR Spent Fuel**




Non-reprocessed spent fuel constitutes a needless waste of recyclable energy; is a 30 times larger volume of waste than the relatively easily managed fission nuclides; and eventually creates a uranium-plutonium ore-body of relatively low radioactivity and of immense potential value as fuel energy. It is the contained plutonium that requires consideration of longer-term management, as this man-made and relatively pure ore body could constitute a proliferation threat if the disposal site (a geological repository) is intentionally breached for any reason other than reprocessing the transuranium contents for re-use as a legitimate source of energy.

## 8. Spent Fuel Interim Storage, Prior to Reprocessing or Disposal

Spent fuel removed from any reactor after it has achieved a significant burn-up, is both highly radioactive and a rapidly diminishing source of heat.

There is a choice of two main processes following discharge of spent fuel. One of these does not consider the immediate possibility of re-processing and the other does. The waste volume and waste management implications following this choice are significantly different.

## 8.1 Disposal

If reprocessing is not chosen as an option, then after a period of from 7 to 10 years of cooling in a water-filled storage bay, the discharged fuel has cooled to just a few hundred watts per tonne, up to about 1 kW per tonne depending upon burn-up. It has also radioactively decayed sufficiently that it may be safely transferred in shielded flasks, to be dry-stored in monitored surface concrete silos or canisters for up to 50 to 100 years at each reactor site where this interim storage option is approved. During this interval, a political/regulatory decision may be made concerning final disposal.

These decisions may:

- extend the duration of operation of this surface storage option;
- revisit the decision about re-processing; or
- direct the spent fuel to a permanent geological deep disposal facility.

## 8.2 Reprocessing

If reprocessing is an option, then the spent fuel is resident in the spent fuel bay for long enough (about 5 months) that it is sufficiently cool that it may be safely transported to a re-processing facility where it is digested in acids prior to selective separation of the fission nuclides from the unburned fuel (including transuranium nuclides) constituents. The less time that the fuel spends in storage the more valuable it is from the fuel value point of view, as some of the shorter half-life fissile or fissionable transuranium nuclides are still present.

The recovered fuel constituents - making up about 95 percent of the spent fuel - are recycled into the reactor, and the fission wastes are conditioned, solidified, and stored, pending removal to permanent non-retrievable disposal. The amount of low level waste produced by reprocessing at the present time, apart from the high level fission waste, is estimated to amount to a cumulative world total by the year 2000 of about 15 000 $m^3$.

## 8.3 Dry Storage of Spent Fuel

Dry Storage consists of specially designed concrete structures with walls typically of 1 meter-thick re-enforced concrete. These may be rectangular or cylindrical structures. They are designed to be weather resistant and physically resistant to upsets due to extreme weather, earthquakes, unauthorized intrusion or sabotage. Once filled, the canisters are secured with tamper-proof seals placed by the IAEA. Safeguards usually include welded steel closures once the vaults or silos have been filled, along with massive concrete closure plugs. Security also involves round-the-clock monitoring of the site and frequent security inspections. Other monitoring of contents takes place on individual vaults and silos by way of drainage lines to ensure no deterioration or leakage of the contents and no significant water ingress. Other site environmental radiation monitoring may include regular radiation surveys by personnel; the placement of thermo-luminescent

dosimeters around the facility; and air and groundwater monitoring in both surface run-off and in deep wells in and around the facility.

None of the hundreds of facilities in existence as Dry-Fuel-Storage locations at commercial facilities to the present has ever shown any significant sign of either deterioration or upset, nor has there been any recorded attempt made to clandestinely access or damage any of the facilities. Any attempt to illegally breach this facility without extensive and expensive preparation and obvious safeguards would, at least in the first few years, result in the likely fatality from radiation effects, of those making the attempt. Any theft of spent fuel would be also be readily and remotely tracked by its easily detected radiation signal by sensitive instruments, increasingly to be found throughout society.

In the case of the CANDU-6 heavy water reactor (600+ MW(e)) reactor, spent fuel - consisting of about 4000 fuel bundles (about 80 tonnes) for each year or so of full power operation - the inventory of monitored and individually identified and tracked spent fuel bundles is kept in the water-filled spent fuel bay for about 7 years. After that cooling time, it can be transferred in shielded flasks to specific locations in the concrete canisters in a secure Dry Fuel Storage facility at each of the reactor sites. All such verified transfers and placements of spent fuel (monitored for validity in the spent fuel bay, by assessing the Cerenkov emissions with sensitive instruments to ensure that dummy or fresh fuel is not substituted) take place under the supervision of an IAEA international inspector. Although the blue Cerenkov 'glow' is readily visible to the naked eye in freshly discharged high burn-up fuel in a water-filled spent fuel bay, it continues as a detectable signal - though not visible to the human eye - for many years. The IAEA inspector places interim seals on the dry-storage containers until they have been filled, and then affixes a 'tamper-proof' seal once the individual canisters have been filled, welded closed, and 'permanently' sealed.

The radioactive decay and decay heat of a typical CANDU natural uranium fuel bundle immediately following discharge at full power, after burn-up of about 7800 MWdays tonne$^{-1}$ of U, and over the longer term are shown in Figure 4 and Table 12.

After this burn-up, the starting content of uranium-235 (0.72 percent is reduced to about 0.22 percent) and plutonium isotopes make up a total of about 0.4 percent of the fuel mass. The decay heat is a direct function of the total burnup that the fuel has experienced and the length of time since the fuel was discharged from the reactor. The heat and radiation output from spent PWR fuel that has reached a burnup of about 30 000 MWdays tonne$^{-1}$ will be about 4 times greater than that of the fuel shown in Table 12.

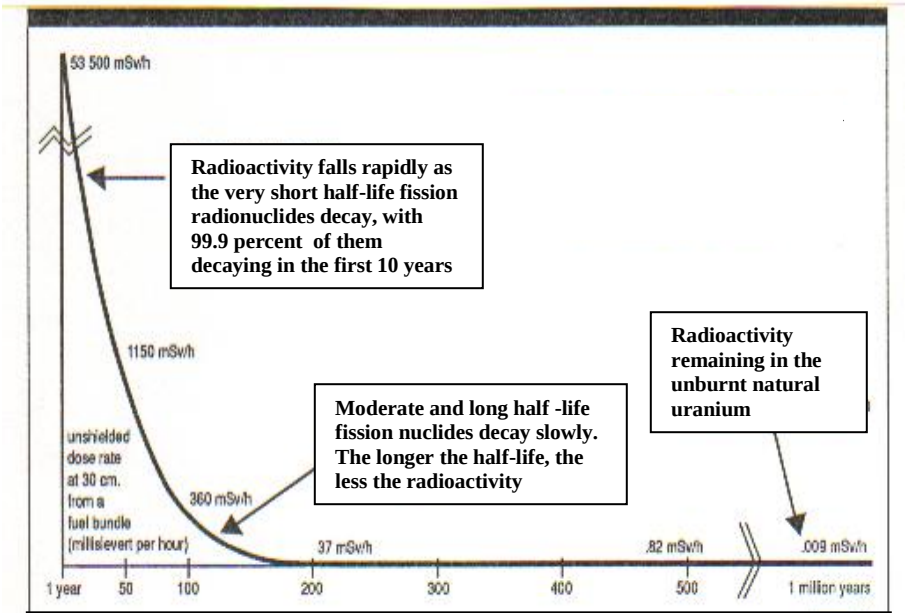



**Figure 4. Radioactive Decay in a CANDU Spent Fuel Bundle of about 21 kilograms of Uranium Oxide. Atomic Energy of Canada Ltd.**

**Table 12. Decay Heat Character in a CANDU Natural Uranium Fuel Bundle vs. Time Since Discharge from a CANDU-600 Reactor (Most Data are from Atomic Energy of Canada Ltd.)**

| Cooling time following discharge | Heat from actinides (watts/bundle -containing 21.0 kg $UO_2$ at the start) | Heat from fission nuclides (watts $bundle^{-1}$) | Total heat (watts/bundle) (burn-up 7800 MWd $Mg^{-1}$ U) |
|---|---|---|---|
| | | | |
| 1 second | 1810 | 23,700 | 25 500 |
| 1 hour | | | 9000 |
| 1day | | | 3000 |
| 1 year | | | 60 |
| 6 y | 0.44 | 5.64 | 6 (300 watts/Mg) |
| 8 y | 0.47 + | 4.44 | 4.9 |
| 10 y | 0.50 + (23.8 watts/Mg) | 3.95 (188 watts/Mg) | 4.4* (209 watts/Mg) |
| 15 y | 0.56 + | 3.34 | 3.9 |
| 20 y | 0.60 + | 2.94 | 3.5 |
| 30 y | 0.66 + | 2.30 | 3.0 |
| 50 y | 0.71 + | 1.43 | 2.1 |
| 100 y | 0.70 | 0.44 | 1.1 (52 watts/Mg) |
| | | | |
| Natural Uranium | | | 0.1 watt/Mg |

**+ Increasing heat in the medium term is from in-growing daughter radionuclides.**

*** About 90 percent of this much-diminished heat output after 10 years, comes from Sr-90 (+Y-90) and Cs-137.**

**For PWR spent fuel with higher burnup, the heat output is about 1 kW $tonne^{-1}$ after ten years.**

## 9. Fuel Reprocessing, Fuel Re-cycling and Advanced Reactors

### 9.1 Fuel reprocessing.

This requires the chemical digestion of spent enriched fuel and returning the chemically separated fuel components (95 to 99 percent unused uranium and transuranium nuclides) into the reactor cycle once they have been re-fabricated as fuel. Compared with the 'wastes' from the 'once-through' cycle, reprocessing reduces the initial volume of materials that need to be managed as High Level Waste, by about 97 percent.

### 9.2 Fuel re-cycling.

Once-through spent fuel from the PWR reactor still contains significant fissile fuel constituents that can be transferred - without chemical reprocessing - to another reactor design (e.g., CANDU) for a second 'once-through' pass. The spent fuel from the first pass achieves additional burn-up and energy output in the second cycle. The additional burn-up renders the spent fuel less economically attractive for immediate reprocessing. Discharged re-cycled fuel is stored pending either permanent disposal or an alternative option, which might include re-processing in the longer term. The spent fuel transferred from the PWR to the second reactor is, of course, mostly eliminated from the PWR waste stream with the exception of re-fabrication wastes, such as fuel cladding.

Both processes recycle spent fuel. The first process is of continuous recycling - at least as far as that is possible - while the second is of just one stage of recycling without consideration of reprocessing.

The amount of electrical energy derived from the use of 1 kilogram of natural uranium in the 'once-through' cycle is about 50 000 kWh (once-through enriched-fuel produces about 250 000 kWh). With reprocessing in the 'closed cycle', the amount of electrical power which can be derived from the same 1 kg of uranium by fully utilizing the uranium-238 and plutonium isotopes, is about 3 500 000 kWh, or about 70 times more than from 'once-through' natural uranium. One kilogram of fully-utilized plutonium is equivalent to about 5 000 000 kWh of electrical power.

The potential energy residing in the depleted uranium world stockpile (about 99.7 percent uranium-238), estimated at about 1.45 million tonnes to the end of 2002, is about 5 million TWh or about 330 times more than the entire world annual production of electricity from all energy sources (estimated at about 15 000 TWh, by 2002), with minimal wastes of any kind and no significant air pollution. It is also comparable to the entire world cumulative production of energy from all sources up to the present time.

Depending upon the burn-up achieved, spent enriched fuel contains about 95 percent U-238, as well as about 1 percent U-235 that has not fissioned; about 1 percent plutonium isotopes produced from U-238 in the fuel (all of relatively low radioactivity); and about 3 percent of highly radioactive fission nuclides.

So far, more than 75 000 tonnes of spent fuel from commercial power reactors have been reprocessed in the world, and annual world reprocessing capacity is now some 5000 tonnes per year, equivalent to about one third of the entire mass of discharged spent fuel. The world commercial reprocessing facilities and their capacities are shown in Table 13.

| **Table 13. World Commercial Reprocessing Capacity (Tonnes per year)** | | |
|---|---|---|
| | | |
| Light water reactor fuel: | France, La Hague<br>UK, Sellafield (THORP)<br>Russia, Chelyabinsk (Mayak)<br>Japan (Rokkasho)<br>**Total LWR** | 1600<br>850<br>400<br>90<br>**2940** |
| | | |
| Other nuclear fuels: | UK, Sellafield<br>France, Marcoule<br>India (Tarapur, Kalpakkam, Trombay)<br>**Total - other** | 1500<br>400<br>200<br>**2100** |
| | | |
| **Total civil capacity** | | **5040** |
| Sources: OECD/NEA 1999 Nuclear Energy Data, Nuclear Engineering International. | | |

**The recovered uranium** may be handled in a normal fuel fabrication plant and blended with low enriched fuel to achieve the fuel feed composition required by the reactor.

**The recovered plutonium** is recycled through a mixed oxide (MOX) fuel fabrication cycle and blended with uranium, usually at the same reprocessing plant that separated it.

MOX fuel is currently being used in commercial nuclear power reactors in Belgium, France, Germany, Japan, Switzerland, and the United Kingdom. It constitutes about 2 percent of all new fuel loading and is steadily increasing. It is being examined for use in the CANDU, U.S. and Russian reactors as a means of economically and safely consuming retired plutonium weapons.

Since about 1963, about 2800 kilograms of re-processed plutonium, contained in about 400 tonnes of MOX fuel, have been consumed in this way. More than 30 European reactors are currently licensed to use MOX fuel for up to about one third of the reactor core load. Future plans are to increase the MOX component up to about one half of the core fuel-load.

### 9.3 Reprocessing and the Closed Fuel Cycle.

The development of a nuclear industry in most countries was initially based upon closing the fuel cycle with reprocessing, fuel re-cycling, and responsible waste management and disposal. The industry also recognized that any nuclear weapons proliferation risk from the reprocessing cycle was unlikely, and could be guarded against. It would also be much less of an ongoing threat to world security than the constant threat of war over political

manipulation, pricing, and shortages of conventional and politically-sensitive energy supplies such as oil.

At the same time, the various anti-nuclear activist groups recognized that if public and political support could be turned away from nuclear power, the reprocessing option in the nuclear power fuel cycle would be abandoned; the cycle could not be closed; and the breeder reactor program would probably be significantly delayed if not stopped. Following this, the continuation of nuclear power as an energy resource would be at least severely limited, if not - they hoped - ended.

### *9.3.1 The Politics of Reprocessing.*

The reprocessing option was dropped in the U.S. in 1977, following deliberation by then President Carter. This purely political decision was supposedly based upon diplomatic concerns about world security and plutonium proliferation, whose risks were assumed to be likely to be significantly augmented following the planned and logical development of the next-generation Fast Breeder Reactor program.

The presumption that the Fast Breeder Reactor would be used for plutonium production and would be likely to increase proliferation risks was shortsighted and unfounded. The Fast Breeder Reactor, as we know today, would most likely be used initially as a net 'burner' of plutonium to reduce weapons stockpiles and to *reduce* risks of proliferation rather than to increase them. Only following that phase, would the FBR be considered as a fuel 'breeder', with all of the bred fuel being stringently controlled and used only for further commercial energy production at the same site at which it was produced. Loss of control was unlikely ever to occur, considering the nature of spent fuel and the existing controls in its management.

Other countries did not follow the Carter political initiative for various reasons relating to their own political self-interest, energy security and self-sufficiency, and the limited availability of alternative energy options. This was especially true in France and Japan, both of which - unlike the U.S. - were limited by a severe shortage of indigenous fossil fuel energy resources and had no intention of crippling their economic growth or international competitiveness by limiting their ability to meet their critical energy needs. France also had every intention of becoming a nuclear power in its own right.

At the same time, various critics of the political initiative suggested that abandoning re-processing in the U.S., because of proliferation concerns in other countries, would be ineffective as it could not possibly achieve the desired intent.

Government officials involved with these policies did not appear to acknowledge that:

1. The production of nuclear weapons does not require the construction of nuclear power facilities. There were other routes - more easily concealed, cheaper and more reliable (uranium enrichment) - to proliferation, than reprocessing of spent fuel;

2. Those countries intent on reprocessing, or even of clandestinely developing weapons technology, would be unlikely to be dissuaded by the naïve, irrational, un-enforceable, and hitherto weak actions of a foreign power.
3. The successful diversion of plutonium would actually be extremely difficult to achieve without detection, even in any rogue state, and would be almost impossible in any internationally-monitored facility;
4. There would be immense political and social damage to any government or group involved in any attempt at diversion, as the recent events in North Korea (2003) showed;
5. Banning reprocessing would be internationally damaging to the U.S. ability to develop advanced nuclear energy technology and to their ability to influence future nuclear developments in other countries.
6. Dictators would not subscribe to any moral or ethical constraints concerning nuclear weapons unless forced to do so by a position of strength, but would take advantage of perceived weakness and vacillation, as North Korea appears to have done since the early 1990s.

Not only did the U.S. position on reprocessing affect only the U.S. and none of the allies or rogue states, but it hampered the ability of the U.S. to maintain its nuclear technological advantage. It also achieved little in terms of limiting proliferation anywhere as it is far easier to construct a nuclear device by enriching uranium, than by constructing reactors and producing - and trying to separate and purify - one specific isotope of plutonium (Pu-239) from the intimately associated transuranium nuclides and highly radioactive fission nuclides through the reprocessing cycle.

At the same time, without the means of burning plutonium as MOX fuel in the reactor cycle, plutonium stockpiles could only increase at many sites, thus augmenting the proliferation risks. By the time President Reagan re-appraised the decision not to reprocess spent fuel, the costs of re-establishing the program, coupled with the onerous nuclear regulatory and licensing climate and the continuing low cost of uranium, were sufficient to ensure that it would not easily proceed.

Former President Clinton also decided to oppose reprocessing on the grounds of proliferation risks, but also attempted to politically influence the operation of foreign reprocessing facilities especially in the U.K., further fouling any atmosphere of nuclear co-operation. At the same time, political appeasement of the regime in North Korea, ironically negotiated to some degree by former President Carter, led to exactly the kind of proliferation situation that his regime had hoped to avoid.

In the U.S., the nuclear industry was whipsawed in uncertain political and rigidly opposing environmental processes and there seemed little point of industry making long-term plans or commitments to the future of civilian nuclear power until the political and regulatory climate had significantly improved. This is likely to happen when the political penalties arising from succeeding energy crises, as well as the perceived environmental and global climate effects because of the continued expansion of fossil fuel use, begin to outweigh the real social penalties of inaction arising from ignoring the benefits and

realities of expanding nuclear power to displace coal and oil, much of the latter imported from politically unstable regions.

The U.S. policy change in 1977 against spent fuel reprocessing had the following general and specific effects (among others):

- It delayed and possibly curtailed (in the U.S.) the possible transition to the future reactor cycles upon which the developed U.S. nuclear program and its future growth had been based;
- It required the continued, relatively intensive exploitation of uranium ore and expensive enrichment, rather than allowing significant displacement by re-cycling unconsumed uranium and plutonium from spent fuel;
- It shelved (at least temporarily) the eventual exploitation and use of the very large stockpiles of depleted uranium in the breeder cycle;
- It created a build-up of spent fuel at each reactor site. Storage facilities had been built assuming transportation and reprocessing of spent fuel after about 150 days of cooling. This resulted in an unplanned, though remediable, shortage of storage space for spent fuel;
- It significantly increased the costs of waste disposal by boosting long-term waste volumes into a 30-times larger volume of radioactive materials (albeit still relatively small) - 97 percent of which was unburned fuel containing little radioactivity when separated from fission HL wastes, and dictated that it was to be managed as waste, with all of the resulting political overtones;
- It significantly increased the time frame for the management of the larger waste volumes, as the trans-uranium nuclides were not being removed from these 'wastes' for return into the reactor cycle;
- It blocked the development of a breeder reactor cycle and effective use of a large part of the depleted uranium stocks. These currently stand at about 600,000 tonnes in the U.S. in 2002, and are equivalent in energy content to about 780 billion tonnes of coal. The potential energy value in this DU stockpile, in terms of electrical energy, is about 100 trillion dollars, assuming $50/MWh.
- It eliminated the possibility of increasing the useable terrestrial fuel resource by up to about 70 fold from the present economic uranium reserves (by using all of the energy in uranium-238 contained in spent fuel), even without consideration of using thorium as fuel (three times more abundant than uranium) or making the much more abundant marginal uranium deposits (including coal ash, phosphates, and seawater) much more economically attractive for their U-238 energy potential in the FBR cycle, rather than for their U-235 content alone. These additions to the fuel resource base would increase the energy resource by many thousands of times, and boosted the resource outlook by millions of years;
- It created a more onerous and uncertain longer-term plutonium concern by disposing of plutonium in a managed waste site, rather than destroying it in the reactor cycle;
- If continued, it will eventually create a relatively large, very long-lived and strategically vulnerable low radioactivity uranium-plutonium ore-body, rather than a small volume of vitrified and relatively short-lived fission waste at the final disposal site.

- It will prolong the continued dependence of the U.S. upon imported (and domestic) fossil fuel energy sources, with controversial environmental degradation because of toxic fossil fuel wastes, and economic and security-of-supply issues.
- It will increasingly make the U.S. and the rest of the world less, rather than more secure, by forcing them to remain dependant upon off-shore fossil fuel resources.

### *9.3.2 Changes in Fuel Design and Utilization.*

Several changes were made in advanced fuel designs and fuel utilization at U.S. reactors to reduce the negative economic effects of this unanticipated change to abandon reprocessing.

- Greater fuel enrichment (along with operational changes) and fuel burn-up (from 40,000 to 60,000 MWdays) was approached in order to derive the greatest value at the least cost from the existing fuel load, as no credit could be applied against residual uranium-235, fissile plutonium-239, or plutonium-241 discharged from the reactor. This reduced the final amount of waste produced, by delaying the need for fuel replacement until declining reactor performance demanded the change.

- Operating modifications were instituted at reactors which aimed for a better conversion from uranium-238 to plutonium in order to derive maximum energy from the once-through fuel. This required a hardening of the neutron spectrum and other operational changes.

| Table 14. Advantages/Disadvantages of Reprocessing or not Reprocessing Nuclear Spent Fuel | |
|---|---|
| **Reprocessing - Closed Cycle** | **No Reprocessing - Once-Through** |
| **Advantages**<br>Recovers the 97 percent unused fuel and its contained energy for recycling.<br>Recovers and uses plutonium (1 percent of the spent fuel.<br>Recovers and 'destroys' plutonium in MOX fuel.<br>Recovers and uses other transuranium elements in spent fuel.<br>Allows transition to the Fast Breeder cycle of reactor operation.<br>Allows use of the 600,000+ tonnes of stored depleted uranium in the US.<br>Makes available, at least 100 times more energy than by not reprocessing, opens up utilization of lower grade uranium ore deposits, and using thorium as fuel, and reduces the need for uranium enrichment.<br>Separates the 3 - 5 percent by volume of High Level fission-waste from spent fuel.<br>Produces low volumes of fission waste requiring a relatively short, waste-management interval.<br>There is minimal requirement for long-term safety and security considerations.<br>Near-surface storage of some waste is possible. | **Advantages**<br>Re-processing facilities are not required, or reprocessing of enriched fuel **was** specifically prohibited (U.S.).<br>Unburned plutonium and transuranium elements are locked with highly radioactive fission products and cannot be readily accessed in the short term.<br>Diversion and proliferation are unlikely in the short term.<br>A Geological Waste repository becomes a plutonium/uranium ore-body that can be re-mined if desired by future generations.<br><br>In the case of natural-uranium fueled reactors such as CANDU, neither enrichment nor reprocessing facilities are required. |
| **Disadvantages**<br>Requires transportation of spent fuel to a central reprocessing facility.<br>Transportation and reprocessing creates a hypothetical risk of diversion and nuclear proliferation. | **Disadvantages**<br>Without reprocessing, future fuel cycle options are limited.<br>97 percent of the potential energy in U-238 and plutonium is not used (wasted).<br>100 percent of spent fuel becomes classified as 'waste'.<br>There is a relatively large 'waste' volume compared with that produced by re-processing.<br>The 'waste' contains unused plutonium and transuranium elements, creating a proliferation and diversion risk.<br>The spent fuel management interval is significantly lengthened .<br>The waste repository becomes a plutonium/uranium ore-body with very long term security and proliferation risks.<br>There is continued environmental degradation because of the continuing need to use fossil fuels for energy. |

## 9.4 Fuel Recycling.

This process appears to be well suited for use in the CANDU reactor whose operation is based upon natural uranium fuel and heavy water moderation and cooling. The CANDU is characterized by continuous refueling at power with several of the 380 (or more) channels refueled each week of operation, and has the advantage of a very high neutron economy.

Briefly, the CANDU reactor has considerable fuel flexibility within a single core load of about 4,500 relatively small fuel bundles. These can include bundles of slightly different compositions, which can be selectively positioned in the core and re-located or removed as needed to achieve the desired core characteristics. The fuel load can include, or be made up of, blended and low enriched fuels (up to 1.2 percent U-235).

Other possible CANDU fuels and fuel mixes include MOX fuels with re-processed plutonium, down-blended weapons HEU and plutonium, depleted uranium, and thorium. Fuel burn-up could also be increased to above 20,000 MWdays/tonne with minor physical modifications.

Fuel recycling does not require chemical re-processing of spent fuel, but takes advantage of the operational characteristics of the CANDU reactor to take the once-through fuel from the PWR cycle and present it as the fuel charge of a CANDU. In this way, the residual, but still elevated level of uranium-235 and plutonium remaining in the PWR 'spent' fuel can achieve an extended burn-up in the heavy water moderated environment. South Korea and Canada are examining recycling non-reprocessed spent fuel directly from the PWR cycle - DUPIC ('Direct Use of Spent PWR fuel in CANDU') into the CANDU reactor - though with some physical re-arrangement of the spent fuel pellets into a form that is amenable to use in the CANDU fuel channel.

Reprocessing the spent fuel from the non-enriched uranium cycle is not envisaged at this time as the remaining fissile nuclide content (U-235 and TU nuclides) is lower than in PWR spent fuel and the economics does not favor reprocessing of non-enriched spent fuel in the short term.

## 9.5 Advanced Reactors (The Fast Breeder Reactor)

The advantage of a nuclear reactor using fast neutrons was recognized in the early 1940s. The potential advantage of fast reactors over thermal reactors was because excess neutrons would be available which could be used for breeding the immense supplies of fertile nuclides (uranium-238, and thorium-232) - of little immediate energy value in the reactor cycle - into fissile nuclides which, in future fuel loadings, could directly contribute to energy production and further breeding. The fast reactor therefore provided the means by which the enormous world-wide energy reserves contained in uranium-238 (99.3 percent of natural uranium) and thorium-232 - far exceeding those contained in all fossil fuel supplies by thousands of times, and hundreds of times greater than those

contained in uranium-235 (0.7 percent of natural uranium) - could be better utilized, and without significant pollution.

The first U.S. breeder reactor - Clementine - was built in Los Alamos in 1946. The core was only about 15 cm high and 15 cm across and it was cooled with mercury which does not significantly absorb fast neutrons.

In its later years, from 1977, the Shippingport PWR was successfully operated as a Light Water Breeder Reactor using uranium-235 as the initial fissile fuel (driver), and thermal breeding of thorium-232 (blanket) to uranium-233, and uranium-238 to plutonium-239 as continuing fuels. It operated for 5 years, producing power, and finished with about 1.3 percent more fissile fuel than it started with.

Breeders were researched and used in pilot facilities until the political decision was made to abandon fuel re-processing, which effectively stopped the breeder program. They still represent the best future option for optimal energy recovery from uranium as well as thorium, and are the only rational and capable means of providing adequate energy resources for the future as well as avoiding the presumed environmental effects of using fossil fuels in energy production.

'Fast', means that fast neutrons, rather than thermal neutrons, are used to achieve 'conversion' of fertile uranium-238 in a breeder blanket, to fissile nuclides (plutonium-239). 'Breeder' indicates that, depending upon the choice of fuel and how the reactor is operated; reactor fuel for the next and succeeding fuel cycles can be 'bred' in the reactor core, as some of the original fuel load is consumed.

Almost all of the present generation of commercial, research and ship reactors are based upon thermal neutron fissioning of uranium-235. Although there is some fast fissioning of uranium-238, and neutron capture conversion of uranium-238 to plutonium-239 in existing reactors, with the production of up to about 40 percent of the total energy output, there is actually little of the uranium-238 that is converted in this way.

The next generation of reactors - Fast Reactors - will use the massive stockpiles of uranium-238 (depleted uranium) byproduct from the uranium-235 enrichment process, natural uranium ore, or thorium-232 (even more crustally abundant than uranium) as fuels. The forward fuel supply outlook with the adoption of the Fast Breeder Reactor is at least many thousands to millions of years.

Various combinations of fissionable and fertile fuels, including retired nuclear weapons plutonium can be readily consumed in the fast reactor cycle. These fast reactors can also be used to destroy other transuranium nuclides that might otherwise be consigned to nuclear waste and can, at the same time, produce large amounts of thermal energy from them.

The choice of fast reactor design and operation covers many fuel options and operations, ranging from net fuel burning, to a balance between fuel burning and fuel production, to

net fuel production, depending upon choice of operational mode (conversion ratio) and fuel load. With a high conversion ratio in a fertile 'blanket' in and around the fuel elements, more fuel can be bred in the energy producing breeding cycle, than is consumed. One of the major advantages is that very little total fuel is needed for a very high energy production rate (about a tonne and a half annually), and there is thus little requirement to move fuel into the reactor site, or waste products out of it, making security and management a relatively simple operation, ensuring that fuel diversion and the much-feared risks of proliferation cannot take place. Indeed, in complete contrast to the political beliefs and concerns of the President Carter years, one of the most significant advantages of the fast reactor is that it is ideally suited to burn-up and destroy stockpiles of plutonium and to bring the management of such sensitive materials into a totally secure environment where they can be eliminated.

Fast reactors have been researched in the U.S., the former U.S.S.R., the U.K. and France. The early test reactors were followed by demonstration reactors: EBR-2 (U.S.A.), BOR-60 (Russia), Rapsodie (France) and DFR (U.K.) built in the 1950s and 1960s. These, in turn, led on to a new generation of prototype power reactors such as the Phenix (France), the Prototype Fast Reactor (PFR) at Dounreay in the U.K., and the BN-350 (Kazakhstan). Most recently, there were developed full-scale power plants designed to make the transition to commercial fast reactor operation; the Superphenix 1(SPX), France, the BN-800 and 1600 (Russia) and others under development in Japan and Europe. Many of the Fast Reactors developed in various countries since the 1940s, with many still under development, are shown in Table 15.

The expectation that the Fast Breeder Reactor would be widely developed and commercially viable by the beginning of the 21st century as a next generation reactor, has not been realized. The continuing availability of relatively cheap fossil fuels, and the related temporary political uncertainties with the funding of nuclear research and development programs of many countries, continue to hamper the research effort and to delay the transition to advanced reactors. However, the long-term and critical importance of Fast Reactors as a means of ensuring greater energy independence and security for many countries, while reducing their pollution emissions remains unchanged, and will be developed once society becomes truly aware of the social and environmental costs of prolonging the use of fossil fuels and of relying upon politically unstable offshore suppliers of fuels.

| Table 15. Fast Breeder Reactors in the World (2002) | | | | | |
|---|---|---|---|---|---|
| Country | Reactor | Fuel | Type* | MW (thermal) | Operational |
| USA | Clementine | Pu | EFR | 0.025 | 1946-53 |
| | EBR 1 | U | EFR | 1.4 | 1951-63 |
| | EBR 2 | U | EFR | 62.5 | 1963-94 |
| | Fermi 1 | U | EFR | 200 | 1963-72 |
| | SEFOR | Pu U | EFR | 20 | 1969-72 |
| | FFTF | Pu U | EFR | 400 | 1980-94 |
| | CRBRP | Pu U | DPFR | 975 | Cancelled |
| | ALMR | U Pu | DPFR | 840 | 2005 |
| | ALMRc | U Pu | CSFR | 840 | To be determined |
| UK | Dounreay DFR | U | EFR | 60 | 1959-77 |
| | Dounreay PFR | Pu U | DPFR | 650 | 1974-94 |
| | CDFR | Pu U | CSFR to EFR | 3800 | |
| France | Rapsodie | Pu U | EFR | 40 | 1966-82 |
| | Phenix | Pu U | DPFR | 563 | 1973- |
| | Superphenix 1 | Pu U | CSFR | 2990 | 1985-98 |
| | Superphenix 2 | Pu U | CSFR to EFR | 3600 | |
| Germany | KNK 2 | Pu U | EFR | 58 | 1972-91 |
| | SNR-2 | Pu U | CSFR to EFR | 3420 | |
| | SNR 300 | Pu U | DPFR | 762 | Cancelled |
| India | FBTR | Pu U | EFR | 40 | 1985- |
| | PFBR | Pu U | DPFR | 1250 | 2010 |
| Japan | Joyo | Pu U | EFR | 100 | 1977- |
| | Monju | Pu U | DPFR | 714 | 1995-96 |
| | DFBR | Pu U | CSFR | 1600 | To be determined |
| Kazakhstan | BN 350 # | U | DPFR | 750 | 1972-99 |
| Russia | BR 2 | Pu | EFR | 0.1 | 1956-57 |
| | BR 10 | U | EFR | 8 | 1958- |
| | BOR 60 | Pu U | EFR | 65 | 1968- |
| | BN 600 | Pu U | DPFR | 1470 | 1980- |
| | BN 800 | Pu U | CSFR | 2100 | To be determined |
| | BN 1600 | Pu U | CSFR | 4200 | To be determined |
| Italy | PEC | Pu U | EFR | 120 | Cancelled |
| Korea | KALIMER | U | DPFR | 392 | To be determined |
| China | CEFR | Pu U | EFR | 65 | To be determined |
| Europe | EFR | Pu U | CSFR | 3600 | To be determined |

* EFR - Experimental Fast Reactor; DPFR - Demonstration or Prototype Fast Reactor; CSFR - Commercial Scale Fast Reactor.
# 150 MW(thermal) is used for desalination.
*Source: IAEA Fast Reactor Data Base*.

**Bibliography**

Chart of the Nuclides. 1996. Nuclides and Isotopes (fifteenth edition). Lockheed Martin, GE Nuclear Energy. (This is a complete layout of the Segre chart of the stable elements, isotopes, and radio-nuclides. It is an invaluable source of nuclide data).

International Atomic Energy Agency (IAEA). Web site address: www.iaea.org (this United Nations site is a comprehensive source of detailed international nuclear and radiation related information of high quality).

Lamarsh, John R. and Baratta, A. J. 2001. Introduction to Nuclear Engineering. Prentice Hall. (This book is a basic comprehensive text for university engineering students. It provides high quality and fairly detailed information on nuclear physics principles, radiation, and nuclear reactors).

Nuclear Energy International. U.S. based site of Nuclear Energy Information. Web site address: www.nei.org. (This site provides a general overview of nuclear energy information in an easily understood format).

U.S. DOE. Web site address: www.eia.doe.gov (this very large site provides comprehensive factual data on energy use throughout the U.S. with links to numerous sites for specific energy information).

World Nuclear Association. Web site address: www.world-nuclear.org. This London-based site provides recent comprehensive and factual general information on almost everything nuclear in the world at a basic level and provides linkages to numerous other nuclear sites).

Word count excluding tables: 11 600

## Appendix 1 - List of Fission Nuclides

| Fission Yield of Isotopes from U-235, Thermal Neutron Fission, in Percent | | | | | |
|---|---|---|---|---|---|
| percent nuclide | Half-life | percent nuclide | Half-life | percent nuclide | Half-life |
| | | | | | |
| 0.000026 percent | | 0.13 percent | | 2.56 percent | |
| Ni-72 | 2.1 s | Zn-80 | 0.54 s | As-87 | 0.8 s |
| Cu-72 | 6.6 s | Ga-80 | 1.66 s | Se-87 | 5.6 s |
| Zn-72 | 46.5 h | Ge-80 | 29 s | Br-87 | 56 s |
| Ga-72 | 13.95 h | As-80 | 16 s | Kr-87 | 76 m |
| Ge-72 | stable | Se-80 | stable | Rb-87 | stable |
| 0.0001 percent | | 0.20 percent | | 3.58 percent | |
| Ni-73 | 0.9 s | Zn-81 | 0.29 s | Se-88 | 1.5 s |
| Cu-73 | 3.9 s | Ga81 | 1.23 s | Br-88 | 16.4 s |
| Zn-73 | 24 s | Ge-81 | 7.6 s, 7.6 s | Kr-88 | 2.84 h |
| Ga-73 | 4.87 h | As-81 | 33 s | Rb-88 | 17.7 m |
| Ge-73 | stable | Se-81 | 18 m, 57 m | Sr-88 | stable |
| 0.0003 percent | | Br-81 | stable | 4.73 percent | |
| Ni-74 | 1.1 s | 0.32 percent | | Se-89 | 0.4 s |
| Cu-74 | 1.6 s | Ga-82 | 0.6 s | Br-89 | 4.4 s |
| Zn-74 | 96 s | Ge-82 | 4.6 s | Kr-89 | 3.16 m |
| Ga-74 | 8.1 m | As-82 | 19 s, 14 s | Rb-89 | 15.2 m |
| Ge-74 | stable | Se-82 | stable | Sr-89 | 50.52 d |
| 0.0011 percent | | Kr-82 | stable | Y-89 | stable |
| Cu-75 | 1.3 s | 0.536 percent | | 5.8 percent | |
| Zn-75 | 10.2 s | Ga-83 | 0.31 s | Br-90 | 1.9 s |
| Ga-75 | 2.1 m | Ge-83 | 1.9 s | Kr-90 | 32.3 s |
| Ge-75 | 82.8 m | As-83 | 13 s | Rb-90 | 2.6 m, 4.3 m |
| As-75 | stable | Se-83 | 22 m, 17 s | Sr-90 | 29 a |
| 0.0031 percent | | Br-83 | 2.39 h | Y-90 | 64 h, 3.2 h |
| Cu-76 | 0.64 s | Kr-83 | stable | Zr-90 | stable |
| Zn-76 | 5.7 s | 1.00 percent | | 5.83 percent | |
| Ga-76 | 29.1 s | Ge-84 | 1.2 s | Se-91 | 0.3 s |
| Ge-76 | stable | As-84 | 6 s, 0.6 s | Br-91 | 0.5 s |
| 0.008 percent | | Se-84 | 3.3 m | Kr-91 | 8.6 s |
| Cu-77 | 0.47 s | Br-84 | 32 m, 6 m | Rb-91 | 58.4 s |
| Zn-77 | 1.4 s | Kr-84 | stable | Sr-91 | 9.5 h |
| Ga-77 | 13.0 s | 1.310 percent | | Y-91 | 58.5 d, 50 m |
| Ge-77 | 5.3 s, 11.3 h | Ge-85 | 0.54 s | Zr-91 | stable |
| As-77 | 38.8 h | As-85 | 2 s | 6.02 percent | |
| Se-77 | stable | Se-85 | 32 s | Br-92 | 0.36 s |
| 0.021 percent | | Br-85 | 2.87 m | Kr-92 | 1.84 s |
| Zn-78 | 1.5 s | Kr-85 | 10.7 a, 4.5 h | Rb-92 | 4.5 s |
| Ga-78 | 5.09 s | Rb-85 | stable | Sr-92 | 2.71 h |
| Ge-78 | 1.45 h | 1.97 percent | | Y-92 | 3.54 h |
| As-78 | 1.5 h | As-86 | 0.9 s | Zr-92 | stable |
| Se-78 | stable | Se-86 | 15 s | | |
| 0.045 percent | | Br-86 | 55.5 s | | |
| Zn-79 | 1.0 s | Kr-86 | stable | | |
| Ga-79 | 3.0 s | | | | |
| Ge-79 | 42 s, 19 s | | | | |
| As-79 | 9.0 m | | | | |
| Se-79 | 3.9 m, 6E5 a | | | | |
| Br-79 | stable | | | | |

| Fission Yield of isotopes from U-235, thermal neutron fission, in percent | | | | | |
|---|---|---|---|---|---|
| percent nuclide | Half-life | percent nuclide | Half-life | percent nuclide | Half-life |
| | | | | | |
| 6.35 percent | | Nb-98 | 2.8 s, 51 m | Tc-104 | 18.2 m |
| Kr-93 | 1.29 s | Mo-98 | stable | Ru-104 | stable |
| Rb-93 | 5.85 s | 6.1 percent | | 0.96 percent | |
| Sr-93 | 7.8 m | Rb-99 | <0.1 s | Nb-105 | 2 s |
| Y-93 | 10.2 h, 0.8 s | Sr-99 | 0.29 s | Mo-105 | 50 s, 30 s |
| Zr-93 | 1.5E6 a | Y-99 | 1.5 s | Tc-105 | 7.6 m |
| Nb-93 | stable | Zr-99 | 2.1 s | Ru-105 | 4.48 h |
| 6.47 percent | | Nb-99 | 15 s, 2.6 m | Rh-105 | 35.4 h, 45 s |
| Kr-94 | 0.2 s | Mo-99 | 65.94 h | Pd-105 | stable |
| Rb-94 | 2.73 s | Tc-99 | 2E5 a, 6 h | 0.402 percent | |
| Sr-94 | 75 s | Ru-99 | stable | Nb-106 | 1.1 s |
| Y-94 | 18.7 m | 6.29 percent | | Mo-106 | 8.4 s |
| Zr-94 | stable | Rb-100 | <0.1 s | Tc-106 | 36 s |
| 6.50 percent | | Sr-100 | 0.2 s | Ru-106 | 372.6 d |
| Kr-95 | 0.78 s | Y-100 | 0.94 s, 0.5 s | Rh-106 | 29.8 s |
| Rb-95 | 0.38 s | Zr-100 | 7.1 s | Pd-106 | stable |
| Sr-95 | 25 s | Nb-100 | 3.1 s, 1.5 s | 0.146 percent | |
| Y-95 | 10.3 m | Mo-100 | stable | Mo-107 | 3.5 s |
| Zr-95 | 64.03 d | Ru-100 | stable | Tc-107 | 21.2 s |
| Nb-95 | 35 d, 3.61 d | 5.2 percent | | Ru-107 | 3.8 m |
| Mo-95 | stable | Sr-101 | 0.12 s | Rh-107 | 21.7 m |
| 6.3 percent | | Y-101 | 0.43 s | Pd-107 | 6.5E6 a |
| Rb-96 | <0.1 s | Zr-101 | 2 s | Ag-107 | stable |
| Sr-96 | 1.06 s | Nb-101 | 7.1 s | 0.054 percent | |
| Y-96 | 6.2 s, 9.8 s | Mo-101 | 14.6 m | Mo-108 | 1.5 s |
| Zr-96 | stable | Tc-101 | 14.2 m | Tc-108 | 5 s |
| Nb-96 | 23.4 h | Ru-101 | stable | Ru-108 | 4.6 m |
| Mo-96 | stable | 4.30 percent | | Rh-108 | 17 s |
| 6.00 percent | | Rb-102 | 0.04 s | Pd-108 | stable |
| Kr-97 | <0.1 s | Sr-102 | 0.07 s | 0.031 percent | |
| Rb-97 | <0.1 s | Y-102 | 0.9 s | Tc-109 | 1.4 s |
| Sr-97 | 0.44 s | Zr-102 | 2.9 s | Ru-109 | 35 s, 13 s |
| Y-97 | 3.7 s, 1.21 s | Nb-102 | 4.3 s, 1.3 s | Rh-109 | 81 s |
| Zr-97 | 16.8 h | Mo-102 | 11.2 m | Pd-109 | 13.43 h |
| Nb-97 | 73.6 m, 54 s | Tc-102 | 5.3 s, 4.4 m | Ag-109 | stable |
| Mo-97 | stable | Ru-102 | stable | 0.025 percent | |
| 5.79 percent | | 3.03 percent | | Tc-110 | 0.83 s |
| Kr-98 | vs | Zr-103 | 1.3 s | Ru-110 | 15 s |
| Rb-98 | 0.11 s | Nb-103 | 1.5 s | Rh-110 | 29 s, 3.1 s |
| Sr-98 | 0.7 s | Mo-103 | 68 s | Pd-110 | stable |
| Y-98 | 2.0 s, 0.6 s | Tc-103 | 54 s | | |
| Zr-98 | 30.7 s | Ru-103 | 39.42 d | | |
| | | Rh-103 | stable | | |
| | | 1.88 percent | | | |
| | | Zr-104 | 1.2 s | | |
| | | Nb-104 | 4.8 s, 1 s | | |
| | | Mo-104 | 60 s | | |

vs: very short half life

| Fission Yield of isotopes from U-235, thermal neutron fission, in percent | | | | | |
|---|---|---|---|---|---|
| percent nuclide | Half-life | percent nuclide | Half-life | percent nuclide | Half-life |
| | | | | | |
| 0.0174 percent | | Ag-118 | 4s, 2.8 s | 0.059 percent | |
| Tc-111 | 0.3 s | Cd-118 | 50.3 m | Cd-126 | 0.51 s |
| Ru-111 | 1.5 s | In-118 | 5 s | In-126 | 1.5 s, 1.45 s |
| Rh-111 | 11 s | Sn-118 | stable | Sn-126 | 1E5 a |
| Pd-111 | 22 m, 5.5 h | 0.013 percent | | Sb-126 | 12.4 d, 11 s |
| Ag-111 | 7.47 d, 65 s | Ag-119 | 2.1 s | Te-126 | stable |
| Cd-111 | stable | Cd-119 | 2.7 m, 2.2 m | 0.157 percent | |
| 0.013 percent | | In-119 | 2.4 m, 18 m | Cd-127 | 0.4 s |
| Ru-112 | 4.5 s | Sn-119 | stable, 293d | In-127 | 1.1 s, 3.8 s |
| Rh-112 | 0.8 s | 0.013 percent | | Sn-127 | 2.1 h, 4.2 m |
| Pd-112 | 21 h | Ag-120 | 1.2 s, 0.32 s | Sb-127 | 3.84 d |
| Ag-112 | 3.14 h | Cd-120 | 50.8 s | Te-127 | 9.5 h, 109 d |
| Cd-112 | stable | In-120 | 44 s, 3 s | I-127 | stable |
| 0.014 percent | | Sn-120 | stable | 0.35 percent | |
| Ru-113 | 2.7 s | 0.013 percent | | Cd-128 | 0.28 s |
| Rh-113 | 0.9 s | Ag-121 | 0.8 s | In-128 | 0.9 s, 0.8 s |
| Pd-113 | 98 s, 89 s | Cd-121 | 13.5 s, 8 s | Sn-128 | 59.1 m |
| Ag-113 | 5.3 h, 68 s | In-121 | 23 s, 3.9 m | Sb-128 | 9.1 h, 10 m |
| Cd-113 | stable | Sn-121 | 27 h, 55 a | Te-128 | stable |
| 0.0118 percent | | Sb-121 | stable | 0.54 percent | |
| Rh-114 | 1.7 s | 0.015 percent | | Cd-129 | 0.27 s |
| Pd-114 | 2.48 m | Ag-122 | 1.5 s | In-129 | 0.6 s, 1.2 s |
| Ag-114 | 4.5 s | Cd-122 | 5.8 s | Sn-129 | 2.5 m, 6.9 m |
| Cd-114 | stable | In-122 | 1.5 s, 10.1 s | Sb-129 | 4.4 h, 18 m |
| 0.0126 percent | | Sn-122 | stable | Te-129 | 69 m, 33.4 d |
| Rh-115 | 0.99 s | 0.0157 percent | | I-129 | 1.6E7 a |
| Pd-115 | 47 s | Ag-123 | 0.39 s | Xe-129 | stable |
| Ag-115 | 2 m, 18.7 s | Cd-123 | 1.84 s | 1.81 percent | |
| Cd-115 | 53.5 h, 45 d | In-123 | 6 s, 48 s | Cd-130 | 0.2 s |
| In-115 | stable | Sn-123 | 129 d, 40 m | In-130 | 0.5 s, 0.5 s |
| Sn-115 | stable | Sb-123 | stable | Sn-130 | 3.7 m, 1.7 m |
| 0.013 percent | | 0.027 percent | | Sb-130 | 38 m, 6.3 m |
| Rh-116 | 0.9 s | Ag-124 | 0.22 s | Te-130 | stable |
| Pd-116 | 12.7 s | Cd-124 | 0.9 s | 2.89 percent | |
| Ag-116 | 2.68 m, 10 s | In-124 | 3.2 s, 2.4 s | In-131 | 0.3 s |
| Cd-116 | stable | Sn-124 | stable | Sn-131 | 61 s, 39 s |
| 0.013 percent | | 0.034 percent | | Sb-131 | 23 m |
| Pd-117 | 5 s | Cd-125 | 0.6 s | Te-131 | 25 m, 32 h |
| Ag-117 | 73 s, 5.3 s | In-125 | 2.3 s, 12.2 s | I-131 | 8.04 d |
| Cd-117 | 2.49 h, 3.4 h | Sn-125 | 9.6 d, 9.5 m | Xe-131 | stable, 12 d |
| In-117 | 43.1 m 1.9 h | Sb-125 | 2.76 a | 4.31 percent | |
| Sn-117 | stable | Te-125 | stable, 58 d | In-132 | 0.22 s |
| 0.011 percent | | | | Sn-132 | 40 s |
| Pd-118 | 3.1 s | | | Sb-132 | 4.1 m, 3.1 m |
| | | | | Te-132 | 78.2 h |

| Fission Yield of isotopes from U-235, thermal neutron fission, in percent | | | | | |
|---|---|---|---|---|---|
| percent nuclide | Half-life | percent nuclide | Half-life | percent nuclide | Half-life |
| | | | | | |
| I-132 | 2.3 h, 83 m | 6.41 percent | | Ce-144 | 284.4 d |
| Xe-132 | stable | I-139 | 2.3 s | Pr-144 | 17 m, 7.2 m |
| 6.70 percent | | Xe-139 | 40 s | Nd-144 | stable |
| Sn-133 | 1.5 s | Cs-139 | 9.4 m | 3.93 percent | |
| Sb-133 | 2.5 m | Ba-139 | 83.1 m | Xe-145 | 0.9 s |
| Te-133 | 12 m, 55 m | La-139 | stable | Cs-145 | 0.58 s |
| I-133 | 20.8 h, 9 s | 6.22 percent | | Ba-145 | 4 s |
| Xe-133 | 5.2 d, 2.2 d | I-140 | 0.86 s | La-145 | 25 s |
| Cs-133 | stable | Xe-140 | 13.6 s | Ce-145 | 2.9 m |
| 7.87 percent | | Cs-140 | 63.7 s | Pr-145 | 5.98 h |
| Sn-134 | 1.04 s | Ba-140 | 12.76 d | Nd-145 | stable |
| Sb-134 | 10.4 s, 0.8 s | La-140 | 40.28 h | Pm-145 | 17.7 a |
| Te-134 | 42 m | Ce-140 | stable | 3.00 percent | |
| I-134 | 52 m, 3.5 m | 5.8 percent | | Cs-146 | 0.33 s |
| Xe-134 | stable | I-141 | 0.46 s | Ba-146 | 2.2 s |
| 6.54 percent | | Xe-141 | 1.72 s | La-146 | 10 s, 6.2 s |
| Sb-135 | 1.71 s | Cs-141 | 24.9 s | Ce-146 | 13.6 m |
| Te-135 | 19.2 s | Ba-141 | 18.3 m | Pr-146 | 24.1 m |
| I-135 | 6.58 h | La-141 | 3.93 h | Nd-146 | stable |
| Xe-135 | 9.1 h, 15 m | Ce-141 | 32.5 d | 2.25 percent | |
| Cs-135 | 2E6 a, 53 m | Pr-141 | stable | Cs-147 | 0.22 s |
| Ba-135 | stable | 5.85 percent | | Ba-147 | 0.70 s |
| 6.32 percent | | I-142 | 0.2 s | La-147 | 4.1 s |
| Sb-136 | 0.82 s | Xe-142 | 1.2 s | Ce-147 | 56 s |
| Te-136 | 18 s | Cs-142 | 1.8 s | Pr-147 | 13.4 m |
| I-136 | 84 s, 45 s | Ba-142 | 10.7 m | Nd-147 | 10.99 d |
| Xe-136 | stable | La-142 | 92 m | Pm-147 | 2.62 a |
| Ba-136 | stable | Ce-142 | stable | Sm-147 | stable |
| 6.19 percent | | 5.96 percent | | 1.67 percent | |
| Te-137 | 4 s | Xe-143 | 0.3 s, 0.96 s | Cs-148 | 0.13 s |
| I-137 | 24.5 s | Cs-143 | 1.78 s | Ba-148 | 0.47 s |
| Xe-137 | 3.84 m | Ba-143 | 15 s | La-148 | 2.6 s |
| Cs-137 | 30.07 a | La-143 | 14.1 m | Ce-148 | 48 s |
| Ba-137 | stable, 2 m | Ce-143 | 33.0 h | Pr-148 | 2.3 m, 2 m |
| 6.77 percent | | Pr-143 | 13.58 d | Nd-148 | stable |
| Te-138 | 1.6 s | Nd-143 | stable | Pm-148 | 5.4 d, 41 d |
| I-138 | 6.4 s | 5.50 percent | | Sm-148 | stable |
| Xe-138 | 14.1 m | Xe-144 | 1.2 s | 1.08 percent | |
| Cs-138 | 32 m, 2.9 m | Cs-144 | 1 s | Ba-149 | 0.34 s |
| Ba-138 | stable | Ba-144 | 11.5 s | La-149 | 1.2 s |
| | | La-144 | 40 s | Ce-149 | 5.2 s |
| | | | | Pr-149 | 2.3 m |

| Fission Yield of isotopes from U-235, thermal neutron fission, in percent | | | | | |
|---|---|---|---|---|---|
| percent nuclide | Half-life | percent nuclide | Half-life | percent nuclide | Half-life |
| | | | | | |
| Nd-149 | 1.73 h | 0.158 percent | | Gd-157 | stable |
| Pm-149 | 53.1 h | Pr-153 | 4.3 s | 0.0033 percent | |
| Sm-149 | stable | Pm-153 | 5.4 m | Pm-158 | 5 s |
| 0.653 percent | | Sm-153 | 46.7 h | Sm-158 | 5.5 m |
| Ce-150 | 4.4 s | Eu-153 | stable | Eu-158 | 45.9 m |
| Pr-150 | 6.2 s | 0.074 percent | | Gd-158 | stable |
| Nd-150 | stable | Pr-154 | 2.3 s | 0.0010 percent | |
| Pm-150 | 2.69 h | Nd-154 | 40 s | Sm-159 | 11.4 s |
| Sm-150 | stable | Pm-154 | 2.7 m | Eu-159 | 18 m |
| 0.419 percent | | Sm-154 | stable | Gd-159 | 18.6 h |
| Ce-151 | 1 s | 0.032 percent | | Tb-159 | stable |
| Pr-151 | 4 s | Nd-155 | 8.9 s | 0.0003 percent | |
| Nd-151 | 12.4 m | Pm-155 | 48 s | Sm-160 | 9.6 s |
| Pm-151 | 28.4 h | Sm-155 | 22.2 m | Eu-160 | 53 s |
| Sm-151 | 90 a | Eu-155 | 4.75 a | Gd-160 | stable |
| Eu-151 | stable | Gd-155 | stable | 0.000085 percent | |
| 0.267 percent | | 0.0149 percent | | Eu-161 | 26 s |
| Ce-152 | 1.4 s | Nd-156 | 5.5 s | Gd-161 | 3.7 m |
| Pr-152 | 3.2 s | Pm-156 | 26.7 s | Tb-161 | 6.91 d |
| Nd-152 | 11.4 m | Sm-156 | 9.4 h | Dy-161 | stable |
| Pm152 | 4.1 m, 15 m | Eu-156 | 15.2 d | 0.00002 percent | |
| Sm-152 | stable | Gd-156 | stable | Eu-162 | 11 s |
| | | 0.0062 percent | | Gd-162 | 8.4 m |
| | | Pm-157 | 10.9 s | Tb-162 | 7.6 m |
| | | Sm-157 | 8.1 m | Dy-162 | stable |
| | | Eu-157 | 15.15 h | others | |

Abbreviations s,m,h,a: seconds, minutes, hours, years.

The total percentage of all fission nuclides in this table is 200 percent. The mass numbers from 80 to about 120 form half of the total, and those from about 120 to 162 make up the remainder.

The individual percentages in the table, refer to the total in the individual series. For example, the total abundance of Pm-157, Sm-157 and Eu-157 is 0.0062 percent. Individual isotope abundance are constantly changing from the moment of their formation, through removal by radioactive decay and for most, by addition from radioactive decay of parent nuclides.

Most data are from the Chart of the Nuclides - Nuclides and Isotopes, fifteenth edition.

**Appendix 2. Graph of Fission Yields from the Thermal Fission of Uranium-235 (from Chart of the Nuclides - Nuclides and Isotopes).**